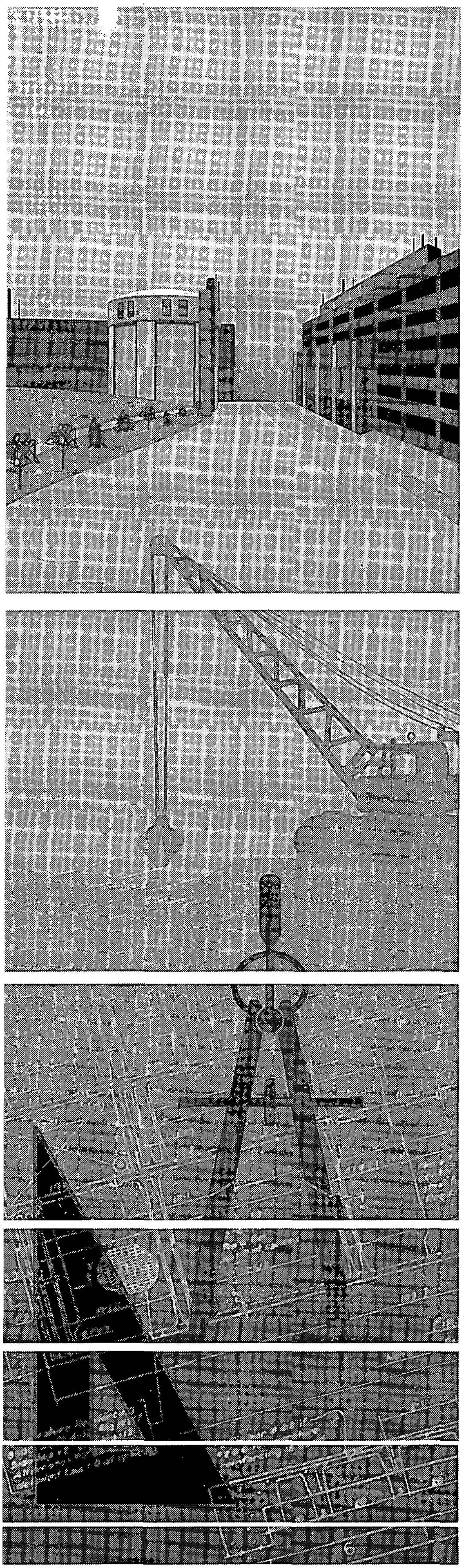

Final Safety Evaluation Report

Related to the Certification of the System 80+ Design

Volume 2

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

August 1994

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NUREG–1462
Vol. 2

Final Safety Evaluation Report Related to the Certification of the System 80+ Design Docket No. 52-002

Chapters 15–22 and Appendices

Manuscript Completed: August 1994
Date Published: August 1994

Associate Directorate for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

ABSTRACT

This final safety evaluation report (FSER) documents the technical review of the System 80+ standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the System 80+ design was initially submitted by Combustion Engineering, Inc., now Asea Brown Boveri-Combustion Engineering (ABB-CE), in accordance with the procedures of Appendix O of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Later ABB-CE requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45.

System 80+ is a pressurized water reactor with a rated power of 3914 megawatts thermal (MWt) and a design power of 3992 MWt at which accidents are analyzed. Many features of the System 80+ design are similar to those of ABB-CE's System 80 design from which it evolved. The staff approved the System 80 design in NUREG-0852, "Safey Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," dated November 1981, and in supplements thereto. Unique features of the System 80+ design include: a large spherical, steel containment; an in-containment refueling water storage tank; a reactor cavity flooding system, hydrogen ignitors, and a safety depressurization system for severe accident mitigation; a combustion gas turbine for an alternate ac source; and an advanced digitally based control room.

On the basis of its evaluation and independent analyses, the NRC staff concludes that ABB-CE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the System 80+ standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR § 52.53 is provided in Appendix E. A final design approval, issued on the basis of this FSER, does not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to Subpart G of 10 CFR Part 2.

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15 TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

ABB-CE evaluated the ability of the System 80+ design to withstand anticipated operational occurrences (AOOs) and a broad spectrum of postulated accidents without posing undue hazard to the health and safety of the public. ABB-CE used the results of these analyses in the CESSAR-DC to show conformance with General Design Criteria (GDC) 10 and 15.

The staff reviewed ABB-CE's transient and accident analyses in the CESSAR-DC for the System 80+ design in accordance with Chapter 15 of the standard review plan (SRP) (NUREG-0800).

The initiating events were assigned to the following seven categories in accordance with SRP Chapter 15:

- increased heat removal by secondary system
- decreased heat removal by secondary system
- decreased reactor coolant flow
- reactivity and power distribution anomalies
- increase in reactor coolant system (RCS) inventory
- decrease in RCS inventory
- radioactive release from a subsystem or component

Initial conditions for the safety analyses are given in Table 15.1 in this chapter. This range of initial conditions is compatible with monitoring functions of the core operating limit supervisory system (COLSS), which is used to aid the operator in maintaining the plant within the limiting conditions for operation (LCOs). COLSS monitoring and calculational functions include peak linear heat rate, margin to departure from nucleate boiling (DNB), total core power, and azimuthal tilt. The COLSS compares these values to their LCOs, and sends an alarm to the operator through the plant computer if an LCO is approached or exceeded, and guides operator actions as is required by the technical specifications (TS) in CESSAR-DC Chapter 16.

A range of fuel parameters based on the first core and future cycles was used for the safety analyses. In the draft safety evaluation report (DSER), the staff noted that ABB-CE did not describe specific parameters selected for each event and required ABB-CE to discuss the Doppler reactivity feedback (DRF) functions, moderator temperature coefficients (MTCs), and the control rod worths used for each of the transient and accident analyses presented in CESSAR-DC Section 6.3.3 and CESSAR-DC Chapter 15. The staff also required ABB-CE to justify the conservatism of the values selected for each event for the first cycle and future cycles. This was designated as DSER Open Item 15.1-1.

In response to the staff's request, ABB-CE submitted CESSAR-DC Sections 6.3.3 and 15.0.3.3 (Amendment R) on November 24, 1992, describing specific parameters used for each event. The parameters include the DRF functions, moderator temperature coefficients (MTCs), and control rod worths used for each of the transients and accidents presented in CESSAR-DC Section 6.3.3 and Chapter 15. The conservative values were assumed in the transient and accident analyses. For the cooldown events resulting from the increased heat removal by secondary systems, the most negative MTC is used to maximize positive reactivity addition due to decreasing coolant temperature. For heatup events resulting from decreased heat removal by secondary systems and decreased RCS flow, ABB-CE used the least negative MTC to minimize the negative reactivity feedback for moderator heatup. For all the transients in CESSAR-DC Chapter 15, ABB-CE assumed that the most reactive control rod is stuck out. These assumptions will increase peak power and heat flux, and decrease DNB ratios (DNBRs). Since conservative values were used in the transient and accident analyses and were appropriately incorporated in CESSAR-DC Sections 6.3.3 and 15.0.3.3 (Amendment R), the staff concludes that DSER Open Item 15.1-1 is resolved.

In Table 15.2, the staff listed the reactor protection system (RPS) trips for which credit was taken in the analyses. The response time for trips was included in the analyses.

The analysis methods used for transient and accident analyses are normally reviewed on a generic basis. ABB-CE topical reports describing analytical methods and the associated NRC approval letters (as stated in the response to the staff request for additional information (RAI) Q440.89) were incorporated into CESSAR-DC Sections 6.3.3 and 15.0.3 and are listed in Table 15.3 of this chapter. The approved methods for non-LOCA analysis include the following computer codes (see Table 15.4):

- CESEC-III (CENPD-107; LD-82-001): Calculates such system parameters as core power, flow, pressure, temperature and valve actions during a transient.

- TORC (CENPD-161) and CETOP (CENPD-206-P-A): TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative channel axial flow that is used to calibrate CETOP. TORC or CETOP is used for DNBR calculations using the CE-1 critical heat flux correlation.

- HERMITE (CENPD-188-A): HERMITE is used to determine short-term response of the reactor core

Table 15.1 Initial conditions considered in the safety analyses

Parameter	Unit of Measurement	Range
Core power	% of 3,914 MWt	0-102
Axial shape index (ASI)		$-0.3 \leq ASI \leq 0.3$
Reactor vessel inlet coolant flow rate	% of 1.69 E+6 L/min (445,600 gpm)	95-116
Pressurizer water level	% of distance between upper tape and lower tap above lower tap	26-60
Core inlet coolant temperature		
0-90% power 90%-100%power	°C (°F) °C (°F)	284-294 (543-561) 288-294 (550-561)
Pressurizer pressure	10^4 kPa (psia)	1.5-1.6 (2,175-2,325)
Steam generator water level		
Low	% of wide range	33.7 (SLB* and FLB* only) 40.7
High	% of narrow range	95.0

*SLB = steamline break, FLB = feedwater-line break

during the postulated reactor coolant pump rotor-seizure event and total loss of flow event.

- COAST (SSAR; CENPD-98): Calculates the time-dependent reactor coolant mass flow rate in each loop during RCP coastdown transients.

- STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 & 4): Calculates the cladding and fuel temperatures for an average or hot fuel rod.

The approved codes for loss-of-coolant accident (LOCA) analyses are discussed as follows:

- CEFLASH-4A (CENPD-133; CENPD-133 Supps. 2, 4-P & 5-P) and CEFLASH-4AS (CENPD-133, Supp. 1): CEFLASH-4A and CEFLASH-4AS determine the primary system hydraulic parameters during the blowdown phase for the analysis of a large-break LOCA and a small-break LOCA, respectively.

- COMPERC-II (CENPD-134; CENPD-134 Supps. 1 & 2): COMPERC-II calculates the system behavior during the refill and reflood phases of a LOCA.

- STRIKIN-II (CENPD-135; CENPD-135 Supps. 2 & 4): It calculates the peak cladding temperature and maximum cladding oxidation.

- FATES3 (CENPD-135; CENPD-135 Supps. 2 & 4; CENPD-139-A; CEN-161(B)-P-A; CEN-161(B)-P-A Supp. 1-P-A), PARCH (CENPD-138, CENPD-138 Supps. 1 & 2), HCROSS (CLD-81-095), and COMZIRC (CENPD-135; CENPD-135 Supps. 2 & 4; CENPD-134; CENPD-134 Supps. 1 & 2): FATES3 calculates the fuel gap conductivity. PARCH and HCROSS determine the steam cooling heat transfer coefficient and steam flow rates during reflood phase, respectively. COMZIRC calculates the core-wide cladding oxidation.

Since the methods used for transient and accident analyses had been approved by the NRC, and since the RCS and core design parameters for the System 80+ design are within the applicable ranges of the approved methods, the staff concludes that the analytical methods used for LOCA analyses in CESSAR-DC Section 6.3.3 and transient analyses in CESSAR-DC Chapter 15 are acceptable.

However, in CESSAR-DC Section 15.0.4, ABB-CE states that for Chapter 15 design-basis events resulting in a

Table 15.2 Reactor protection system (RPS) trips used in the safety analyses

Event	RPS	Analysis Setpoint
Feedwater and steamline breaks	High pressurizer pressure	1.7 E+4 kPa (2,475 psia)
	Low pressurizer pressure	1.07 E+4 kPa (1.555 psia)
	Low steam generator pressure	4.96 E+3 kPa (719 psia)
	Low steam generator water level	33.7% wide range
	High steam generator water level	95% wide range
	*CPC low RCP shaft speed	95%
	*CPC variable overpower	115%
Other events	High logarithmic power level	0.05%
	Variable overpower	119%
	*CPC variable overpower	115%
	High pressurizer pressure	1.68 E+4 kPa (2,434 psia)
	Low pressurizer pressure	1.18 E+4 kPa (1,705 psia)
	Low steam generator pressure	5.38 E+3 kPa (781 psia)
	Low steam generator water level	40.5% wide range
	High steam generator water level	95% narrow range
	Steam generator low flow	80% hot leg flow
	*CPC low RCP shaft speed	95%
	*CPC coincident low pressure/DNBR	1.39 E+4 kPa (2,015 psia) (credited for **LLB and SGTR)

*Core protection calculator
**LLB = letdown line break, SGTR = steam generator tube rupture

violation of the DNBR safety limit (e.g., 1.24), the statistical convolution method was used to calculate the number of failed fuel rods. This method assigned a probability of occurrence of DNB as a function of DNBR. In the meetings held on November 5, November 16, and December 9, 1992, and in a submittal dated December 18, 1992, ABB-CE stated that the NRC staff had approved its use of the convolution method (CENPD-183-A) and requested that the NRC staff permit it to continue to use this method to determine the number of failed fuel rods for System 80+ accident analyses. The staff finds that even though the convolution method has been approved for generic applications to locked rotor events and the plant-specific application (Palo Verde) for control rod ejection accidents, its application, as implied by the method described in CENPD-183-A, is restricted to the DNB probability distribution for the specific fuel types. The approved convolution method described in CENPD-183-A is based on the CE-1 correlation for DNBR calculations and is applicable to CE 14 x 14 and 16 x 16 fuel types.

In the DSER, the staff asked ABB-CE to justify the adequacy of the convolution method used for failed rod determination for the System 80+ design and analysis. This was designated as DSER Open Item 15.1-2.

In response, ABB-CE submitted additional information in CESSAR-DC Section 15.0.4 (Amendment N) indicating that the DNB probability distribution used in the CESSAR-DC analyses is based on the parameters of the 16 x 16 fuel design and the CE-1 correlation. The staff found that the application of the convolution method to the

Table 15.3 ABB-CE Topical Reports and Letters, and NRC Approval Letters in Response to RAI Q440.89

Combustion Engineering Reports

CEN-161(B)-P-A	"Improvements to Fuel Evaluation Model," August 1989 (proprietary).
CEN-161 Supp. 1-P-A	"Improvements to Fuel Evaluation Model," January 1992.
CENPD-98	"COAST Code Description," April 1973 (proprietary).
CENPD-107	"CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," April 1974 (proprietary).
CENPD-133	"CEFLASH -4A, a FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis," April 1974 (Proprietary).
CENPD-133 Supp. 1	"CEFLASH-4AS, a Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-Coolant Accident," August 1977 (proprietary).
CENPD-133 Supp. 2	"CEFLASH-4A, a FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modification)," February 1975 (proprietary).
CENPD-133 Supp. 4-P	"CEFLASH-4A, a ORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977 (proprietary).
CENPD-133 Supp. 5-P	"CEFLASH-4A, a FORTRAN-77 Digital Computer Program for Reactor Blowdown Analysis," June 1985 (proprietary).
CENPD-134	"COMPERC-II, a Program for Emergency Refill-Reflood of the Core," August 1974 (proprietary).
CENPD-134 Supp. 1	"COMPERC-II, a Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975 (proprietary).
CENPD-134 Supp. 2	"COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985.
CENPD-135	"STRIKIN-II, a Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1974 (proprietary).
CENPD-135 Supp. 1	"STRIKIN-II, a Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)," December 1974 (proprietary).
CENPD-135 Supp. 4	"STRIKIN-II, a Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976 (proprietary).
CENPD-138	"PARCH, a FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974 (proprietary).
CENPD-138 Supp. 1	"PARCH, a FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975 (proprietary).
CENPD-138 Supp. 2	"PARCH, a FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977 (proprietary).
CENPD-139-A	"CE Fuel Evaluation Model," July 1974 (proprietary).
CENPD-161	"TORC Code--A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.
CENPD-183-A	"Loss of Flow: CE Methods for Loss of Flow Analysis," July 1975.

CENPD-188-A	"HERMITE, a Multi-dimensional Space-Time Kinetics Code for PWR Transients," July 1976 (proprietary).
CENPD-190-A	"CE Method for Control Element Assembly Ejection Analysis," January 1976. (NRC approval letter, June 10, 1976)
CENPD-206-P-A	"TORC Code--Verification and Simplified Modeling Methods," June 1981.
CENPD-254-P-A	"Post-LOCA Long Term Cooling Evaluation Model," June 1980.
SSAR	"Standard Safety Analysis Report," CESSAR Docket No. STN-50-470, December 1975.

<u>Combustion Engineering Letters</u>

LD-81-095	"CE ECCS Evaluation Model Flow Blockage Analysis," December 1981.
LD-82-001	"CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," (Enclosure 1-P to letter from A.E. Scherer (CE) to D.G. Eisenhut (NRC), January 1, 1982.

<u>NRC Approval Letters</u>

December 4, 1974, Letter from O.D. Parr (NRC) to F.M. Stern (CE).

June 13, 1975, Letter from O.D. Parr (NRC) to F.M. Stern (CE).

June 10, 1976, Letter from NRC to CE.

November 12, 1976, Letter from K. Kniel (NRC) to A.E. Scherer (CE).

September 27, 1977, Letter from K. Kirsh (NRC) to A.E. Scherer (CE).

April 10, 1978, Letter from K. Kniel (NRC) to A.E. Scherer (CE).

September 6, 1978, Letter from R.L. Baer (NRC) to A.E. Scherer (CE).

December 11, 1980, Letter from NRC to CE.

July 21, 1981, Letter from R.A. Clark (NRC) to W. Cavanaugh III (AP&L).

March 13, 1983, Letter from R.A. Clark (NRC) to A.E. Lundvall, Jr. (BG&E).

April 3, 1984, Letter from C.O. Thomas (NRC) to A.E. Scherer (enclosure).

July 31, 1986, Letter from D.M. Crutchfield (NRC) to A.E. Scherer (CE).

System 80+ design is within the applicable limits (the CE-1 correlation applying to the CE 16 x 16 fuel design) of the approved method and, therefore, the staff concludes that it is acceptable. However, if ABB-CE makes changes to the System 80+ fuel design such that other than a 16 x 16 fuel rod array is used or such that the CE-1 DNBR correlation is invalidated, ABB-CE is required to submit technical justification demonstrating the acceptability of the convolution method in the fuel failure calculation. On this basis, DSER Open Item 15.1-2 is resolved.

In the DSER, the staff stated that ABB-CE had not identified all System 80+ design features that deviate from the requirements of the EPRI URD. ABB-CE should have revised the design deviation list that was sent to the NRC in a letter dated August 28, 1990. The revision should listall design deviations and should justify the adequacy of the deviations for System 80+. This was designated as DSER Open Item 15.1-3. The staff has reviewed ABB-CE's responses addressing the System 80+ design deviations from the EPRI URD requirements and found that the responses are acceptable for closure of DSER Open Item 15.1-3. The staff's evaluation for the closure of DSER Open Item 15.1-3 appears in Section 1.1 of this report.

In the original submittal (Amendment H of CESSAR-DC), ABB-CE requested a 3-second delay time for a

Table 15.4 Computer codes used in the safety analyses

Analysis and Code	Documentation	Date of NRC Approval Letter
Non-LOCA Analysis		
CESEC-III	LD-82-001	April 3, 1984
CETOP-D (Rev. 1)	CENPD-161, CEN-160-S-P	July 21, 1981
TORC (thermal-hydraulic code)	CENPD-206-P-A	December 11, 1980
HERMITE (neutronic code)	CENPD-188-A	June 10, 1976
COAST (RCS pump flow cooldown code)	CENPD-98	December 4, 1974
STRIKIN-II (fuel behavior code)	CENPD-135, Supplements 2 and 4	June 10, 1976
Small-break LOCA Analysis		
CEFLASH-4AS (systems blowdown code)	CENDF-133, Supplements 1 and 4	June 13, 1975; September 27, 1977
COMPERC-II (reflood system code)	CENPD-134, Supplements 1 and 2	September 27, 1977; July 31, 1986
STRIKIN-II (fuel behavior code)	CENPD-135, Supplements 2, 4, and 5	June 13, 1975; November 12, 1976; September 6, 1978
FATES3 (fuel gap conductivity code)	CENPD-139-A, CEN-161(B)	December 4, 1974; March 31, 1983
PARCH (pool boiling heat transfer code)	CENPD-138, Supplements 1 and 2	June 13, 1975; April 10, 1978
Large-break LOCA Analysis		
CEFLASH-4A (system blowdown code)	CENPD-133, Supplements 2, 4, and 5	June 13, 1975; July 31, 1986
COMPERC-II	CENPD-134, Supplements 1 and 2	June 13, 1975; July 31, 1986
STRIKIN-II	CENPD-135, Supplements 2, 4, and 5	June 13, 1975; November 12, 1976; September 6, 1978
FATES3	CENPD-139, CEN-161(B)	December 12, 1974; March 31, 1983
PARCH (steam cooling heat transfer code)	CENPD-138, Supplement 1, LD-81-095	June 13, 1975; July 31, 1986; April 10, 1978
COMZIRC (cladding oxidation code)	CENPD-134, Supplements 1 and 2	June 13, 1975
HCROSS (hydraulic code to determine steam flow rates during reflood phase)	LD-81-095	July 31, 1986

loss-of-offsite power (LOOP) caused by turbine trip. The request is based on the grid stability analysis for the worst-case grid within the United States. At the March 17, 1992, meeting, the staff indicated that additional information was required to justify the 3-second delay and asked ABB-CE to update the grid stability analysis and submit some operational data to validate the evaluation. Should ABB-CE decide to submit adequate supplemental information, the staff will require the following actions:

- Since actual loss of grid will vary, ABB-CE must demonstrate that loss-of-offsite power (LOOP) at any time in excess of 3 seconds will not lead to failure to transfer emergency core cooling system (ECCS) loads. This should account for the possibility that safety injection (SI) pumps may have started on normal ac sources and then lost power as the grid isolated, until the diesel generators (DGs) started and loaded. This could result in SI pumps coasting down, lines draining, and possible vapor binding of pumps. ABB-CE must demonstrate that successful pump restart is assured upon DG reload and is not precluded by steam binding overspeed or water hammer.

- ABB-CE has determined the minimum time for grid isolation from a postulated worst-case grid. However, a grid's installed capacity, demand, and spinning reserve vary over time. ABB-CE should indicate the measures that will be taken to ensure that real-time grid conditions will continue to meet the assumptions inherent in the 3-second LOOP delay.

This was designated as DSER Open Item 15.1-4. By letter dated December 18, 1992, ABB-CE agreed not to take credit for 3-second LOOP delay in the transient and accident analysis. The staff finds that the reanalysis in CESSAR-DC Section 6.3.3 and Chapter 15 (Amendment R) assumed that the loss of offsite power occurred coincident with a turbine trip. This approach is consistent with the one that the NRC previously approved for ABB-CE reactors. On this basis, the staff concludes that DSER Open Item 15.1-4 is resolved.

In the DSER, the staff also had concerns about ABB-CE's application of the GDC to accident analysis. For some transient analyses (such as the loss of condenser vacuum event analysis) reported in the original submittal, ABB-CE did not postulate the unavailability of offsite power as part of the event, but rather took it as an additional single failure to show limiting fault criteria were not exceeded. However, GDC 17 in 10 CFR Part 50, Appendix A requires, in part:

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents.

In accordance with the requirements of GDC 17, a LOOP should not be considered as a single-failure event and should be assumed in the analysis for each event without changing the event category. ABB-CE was required to discuss each of the transient and accident analyses in the CESSAR-DC to justify that the analyses conform to the GDC 17 requirements given above. If the existing analyses did not conform to the GDC 17 requirements, ABB-CE should have reanalyzed the transient and accident analyses in accordance with GDC 17 and should submit the analyses for the staff to review. This was designated as DSER Open Item 15.1-5.

By letter dated December 18, 1992, ABB-CE agreed to comply with GDC 17, which requires that the LOOP not be treated as a single failure. ABB-CE reported included the results of reanalysis in the CESSAR-DC (Amendment R) Section 6.3.3 and Chapter 15. The staff reviewed the submittal and found that ABB-CE considered the LOOP in all the events analyzed and applied the acceptance criteria specified in the related SRP sections for the event with or without LOOP. The staff concludes that ABB-CE's approach complies with the requirements of GDC 17. On this basis, DSER Open Item 15.1-5 is resolved.

After the DSER was published, ABB-CE proposed to increase the rated power by 3 percent from 3,800 megawatts-thermal (MWt) to 3,914 MWt, and reanalyzed the transient and accident events to support the request of the power upgrade for the System 80+ design. To reflect the design changes related to the power upgrade, ABB-CE changed the assumptions used in the original analysis for the following parameters:

(1) The range of initial conditions for the CESSAR-DC Chapter 15 analyses is reduced for pressurizer pressure and core inlet temperature. The revised values are listed in Table 15.1 of this chapter.

(2) The minimum flow rate through the pressurizer safety valves is increased by 14 percent.

(3) The maximum charging flow rate is reduced by 20 percent to 567 L/minute (150 gpm).

(4) The most positive MTC at full power is changed from 0.0 to -1.8×10^{-4} Δ-ρ/°C (0.0 to -0.1×10^{-4} Δ-ρ/°F). At zero power, the MTC is reduced from 0.9×10^{-4} Δ-ρ/°C ($+0.5 \times 10^{-4}$ Δ-ρ/°F) to 0.0.

(5) The CEDM coil delay time is reduced from 0.8 second to 0.5 second.

(6) The 90-percent CEA insertion time is reduced from 3.66 seconds to 3.5 seconds.

(7) The site atmospheric dilution factors, χ/Qs, are changed to the EPRI URD values.

(8) The offsite doses for events involving fuel failure are computed using the NUREG-1465 source term.

The staff reviewed these changes in the assumptions used for the power upgrade reanalysis and finds that they are acceptable for the following reasons:

(1) Item 1 — The range of initial conditions is bounded by the limits specified in TS 3.4.1 (CESSAR-DC Chapter 16).

(2) Item 2 — The flow capacity of the pressurizer safety valves is within the design capacity described in CESSAR-DC Section 5.4.13 and its Appendix 5A.

(3) Item 3 — The flow capacity of the CVCS charging pump is bounded by the design flow described in CESSAR-DC Section 9.3.4.

(4) Item 4 — The values of the MTC are within the limits of TS 3.1.4.

(5) Item 5 — The CEDM coil delay time bounds the limits obtained from data of shop test performed on equipment identical to that of the System 80+ CEDM design. The delay time of 0.5 second also bounds field test data on the Palo Verde reactor (a System 80 plant with a similar CEDM design), that shows a maximum CEDM delay time of 0.49 second. The results of these tests are included in CESSAR-DC Section 15.0.2 (Amendment R).

(6) Item 6 — The 90-percent CEA insertion time of 3.5 seconds is consistent with the measured data for the test described in Figure 4B-4 of CESSAR-DC Appendix 4B and is bounded by TS 3.1.5.

(7) Items 7 and 8 — The assumptions related to χ/Q and the source term for the radiological calculation are consistent with the staff's position. (See the staff's evaluation in Section 15.4 of this chapter.)

ABB-CE used the TORC code instead of the CETOP code to calculate the minimum DNBRs for the feedwater line break, loss of condenser vacuum, locked rotor, and steam generator tube rupture events. Since the staff approved both TORC and CETOP previously for the DNBR calculation (as discussed in this section), this approach is acceptable.

ABB-CE provided the results of reanalysis for the power upgrade in CESSAR-DC Section 6.3.3 and Chapter 15 (Amendment R). The staff reviewed the submittal and its evaluation follows.

15.2 Transient Analyses

For the System 80+ design, ABB-CE analyzed all events described in SRP Chapter 15 and presented the limiting event or event combination for each category in analytical detail in CESSAR-DC Chapter 15. For nonlimiting events, ABB-CE prepared qualitative discussions explaining why the events are not limiting. The staff's evaluation of the system responses and thermal-hydraulic behaviors of the analyzed transients is discussed in this section for transient analyses and in Section 15.3 for accident analyses. The staff's evaluation of the radiological consequences for various postulated events is presented in Section 15.4 of this chapter.

15.2.1 Increase in Heat Removal by the Secondary System

In CESSAR-DC Section 15.1, ABB-CE presented the analytical results of the events with increase in heat removal by the secondary system in accordance with SRP Sections 15.1.1 through 15.1.4. These SRP sections correspond to the following subjects:

- decrease in feedwater temperature (SRP Section 15.1.1)

- increase in feedwater flow (SRP Section 15.1.2)
- increase in steam flow (SRP Section 15.1.3)
- inadvertent opening of a steam generator (SG) relief or safety valve (SRP Section 15.1.4)

ABB-CE's acceptance criteria for moderate-frequency transients discussed in CESSAR-DC Chapter 15 are consistent with the guidelines of SRP Chapter 15. The acceptance criteria are:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressure.
- Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above 95/95 DNBR safety limit.
- A transient of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
- A transient of moderate frequency in combination with single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding.

ABB-CE evaluated these four overcooling event categories and determined that the limiting event is the event of an inadvertent opening of an SG atmospheric dump valve (IOSGADV), which belongs to Category 4 events documented in this section. Since the IOSGADV event results in a higher cooldown which causes a higher power increase and consequently results in the highest DNBR decrease during the transients, ABB-CE determined that the IOSGADV event is the limiting overcooling event.

In the analysis of the IOSGADV events, a maximum steam flow of 11 percent of the total SG design flow was assumed to release from an ADV. With no operator intervention or system malfunctions, the analysis showed that the core power of this event increased and stabilized at 115 percent of the rated core power. To include the maximum cooldown effect, the feedwater control system was assumed to operate in the automatic mode to maximize the feedwater to the SGs. As a result, the SG water level was maintained and an automatic turbine trip would not be predicted to occur. The analytical assumptions and initial conditions were chosen so that the greatest overpower conditions would occur as a result of the increase in steam flow. If the core power increases beyond 115 percent of the rated power, the CPC will initiate a reactor trip. To comply with the GDC requirements, a LOOP was assumed to occur simultaneously with a turbine trip. In the analysis, the operator action was assumed to actuate reactor and turbine trips at 30 minutes after the initiation of the event. The RCPs were assumed to begin coastdown at the time of turbine trip. To limit the steam released to the atmosphere, the ADV was assumed to close at 50 minutes after event initiation. The staff finds that the assumption of delay time of 50 minutes to close ADV is conservative for the radiological release calculation because the staff's position stated in SRP, Chapter 15, allows operator actions to be credited for event mitigation after 30 minutes following initiation of the events.

ABB-CE also assessed the consequence of the limiting single failure for each event. The most limiting single failure identified for the four overcooling event categories discussed in this section is the loss of the feedwater control system (FWCS) reactor trip override (RTO). This fault results in the feedwater control system failing to reduce feedwater flow after reactor trip. The feedwater continues to remove the heat from the RCS at a high rate, thereby reducing RCS pressure, and resulting in a lower DNBR value. The results of ABB-CE's analysis indicate that: the minimum calculated DNBR is 1.30 for IODSGADV with LOOP, and is 1.29 for IOSGADV with LOOP and the most limiting single failure. The calculated peak RCS and SG pressures for both cases are within the safety limits of 110 percent of the design pressures. The IOSGADV event with and without a single failure is the limiting event for the four overcooling event categories. Since it does not result in a minimum DNBR less than the safety DNBR limit of 1.24, ABB-CE concluded, and the staff agrees, that no fuel damage would occur for any of the four overcooling event categories. On the basis of the calculational results showing no violation to the safety pressure limits and safety DNBR limits, the staff concludes that the analysis is acceptable.

15.2.2 Decrease in Heat Removal by the Secondary System

In CESSAR-DC Section 15.2, ABB-CE reported the analytical results for various transients resulting from a decrease in heat removal by the secondary system, and identified the limiting cases for the consideration of integrity of RCS system boundary and fuel rod cladding to withstand the consequences of transients. The following transients were analyzed in accordance with the guidance in SRP Section 15.2:

- loss of external load
- turbine trip
- loss of condenser vacuum
- main steam isolation valve closure
- loss of nonemergency ac power to the station auxiliaries
- loss of normal feedwater flow

ABB-CE's analysis showed that the most limiting case is the loss of condenser vacuum (LOCV) event, which may be caused by the failure of the main condenser evacuation system to remove noncondensable gases, or excessive inleakage of air. Similar to the turbine trip and the loss of load event, the LOCV event also results in a turbine trip. However, feedwater terminates following a LOCV event while it ramps down following the turbine trip and the loss of load event. The larger reduction in heat removal due to sudden termination of feedwater results in a higher peak

RCS pressure and lower minimum DNBR for the LOCV event.

Various combinations of initial core inlet temperature, pressurizer pressure and pressurizer water level were considered in order to select a set of initial conditions to generate the highest peak pressure and lowest minimum DNBR resulting from the LOCV event. ABB-CE's analysis indicated that decreasing the initial core inlet temperature reduces the initial steam generator pressure, thereby delaying the opening of the main steam safety valves and associated heat removal effects. Thus, the initial inlet temperature was assumed at the minimum value of the operating range. The pressurizer pressure and pressurizer water level were chosen to maximize the delay time to trip the reactor, and to open the main steam safety valves, resulting in a maximum peak pressure.

In compliance with GDC 17 that requires a loss of offsite power (LOOP) to be considered in the analysis, ABB-CE assumed that the LOOP occurred coincidently with the initiation of the LOCV event, which results in a simultaneous turbine trip. The LOOP causes the four RCS pumps to coast down, which in turn, results in a reactor trip signal generated by the low pump shaft speed. This reactor trip signal occurs earlier than that generated by the high pressurizer pressure signal from the LOCV event without LOOP. Consequently, ABB-CE indicated that with respect to peak pressure, the LOCV event with power available has a longer reactor trip delay time for the RCS pressure to increase and, thus, is more limiting than the event with LOOP. With respect to fuel performance, the LOCV event without LOOP, results in a lowest minimum DNBR. ABB-CE evaluated the single failures listed in CESSAR-DC Table 15.0-4 and concluded that no single failure will result in a lower minimum DNBR or a higher peak RCS pressure than that for the LOCV event without LOOP following a turbine trip.

ABB-CE's analyses showed that for the limiting LOCV event among the heatup events discussed in this section, the minimum calculated DNBR is 1.26, indicating no fuel failure. The maximum peak RCS pressure is 1.88×10^4 kPa (2726 psia), which is less than 110 percent of the design pressure, thereby assuring integrity of the pressure boundary for any of the heatup events discussed in this section. The staff finds that the results of these analyses are in conformance with the acceptance criteria of SRP Sections 15.2.1 through 15.2.5. Therefore, the staff concludes that the analyses are acceptable. On this basis, DSER Open Item 15.2.2-1 is resolved.

15.2.3 Decrease in Reactor Coolant Flow Rate

A complete loss of forced reactor coolant flow will result from the simultaneous loss of electrical power to all reactor coolant pumps. The only credible failure which can result in a simultaneous loss of power is a complete loss of offsite power. In addition, since a loss of offsite power is assumed to result in a turbine trip and renders the steam dump and bypass system function unavailable, the plant cooldown is performed utilizing the main steam safety valves and atmospheric dump valves.

In CESSAR-DC Section 15.3.1 (Amendment R), ABB-CE presents the analytical results for events involving total loss of forced reactor coolant flow that leads to a decrease in reactor coolant flow. The partial loss of forced reactor coolant flow, resulting in smaller loss in the DNBR margin, is bounded by the total loss of forced reactor coolant flow.

A loss of power to all reactor coolant pumps produces a reduction of coolant flow through the reactor core. The reduction in coolant flow rate causes an increase in the average coolant temperature in the core and a decrease in margin to DNB. A low RCP shaft speed trip is initiated by the core protection calculator to prevent the minimum DNBR calculated with the CE-1 correlation from decreasing below the safety DNBR limit of 1.24 during the transient.

For the loss of offsite power event, the minimum DNBR and maximum system pressure occur during the first several seconds of the transient following the reactor trip by the CPC on low DNBR signal. ABB-CE evaluated possible single failures listed in CESSAR-DC Table 15.0.4 and concluded that none of the single failures will result in a lower minimum DNBR and a higher RCS pressure than that predicted for a total loss of forced reactor coolant flow event resulting from the simultaneous loss of electrical power to reactor coolant pumps. In the analysis, a LOOP was assumed to be coincident with a turbine trip.

ABB-CE's analyses showed that for the limiting loss of forced reactor coolant flow event, the minimum calculated DNBR is 1.27, indicating no fuel failure. The maximum peak steam generator pressure is 8.8×10^3 kPa (1,273 psia) and peak RCS pressure is 1.84×10^4 kPa (2,665 psia), each less than 110 percent of the design pressure, thereby assuring integrity of the pressure boundary. The staff finds that the results of the analyses are in compliance with the acceptance criteria of SRP Section 15.3.1. Therefore, the staff concludes that the analyses are acceptable.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Uncontrolled CEA Withdrawl From a Subcritical or Low Power Condition With Loss of Offsite Power

In the CESSAR-DC, ABB-CE analyzes the consequences of an uncontrolled control element assembly (CEA) withdrawal at low power with a loss of offsite power. The CEA withdrawal at low power with a LOOP was determined to be more limiting than without a LOOP. Such a transient can be caused by a failure in the control element drive mechanism, control element drive mechanism control system, reactor regulating system, or by operator error. The analysis assumes a conservatively small (in absolute magnitude) negative Doppler coefficient and the most positive moderator coefficient. The reactivity insertion rate corresponds to the largest insertion rate expected from the sequential withdrawal of the CEA group from the fully inserted position at the maximum speed of 76.2 cm/min (30 in./min). The transient is terminated with a minimum DNBR greater than 1.24 in the hot channel. Fuel centerline temperatures do not exceed the melting temperature of uranium dioxide (UO_2), and the highest RCS pressure produced is well below the emergency limit of 1.9 x 10^4 kPa (2,750 psia).

The staff reviewed the reactivity worths and reactivity coefficients used in the analysis and concludes that ABB-CE used conservative values. The staff reviewed the calculated consequences of this design transient and concludes that they conform with the acceptance criteria in the SRP and are, therefore, acceptable.

The requirements of GDC 20, that protection be automatically initiated, and GDC 25, that a single failure of the protection system does not result in violation of specified acceptable fuel design limits, have been satisfied.

15.2.4.2 Uncontrolled CEA Withdrawal From Power With Loss of Offsite Power

ABB-CE has analyzed the consequences of an uncontrolled CEA withdrawal in the power operating range. The CEA withdrawal at power with a LOOP was determined to be more limiting than without a LOOP. The LOOP was assumed to be coincident with a turbine trip. The effect of the resulting power transient causes an increasing temperature and pressure transient which, together with the power distribution shift to the top of the core, causes a rapid approach to the fuel design limits. The initial conditions assumed in the analysis include a power level of 102 percent of full power, a top peaked core average axial power distribution, a conservatively small Doppler coefficient, and the most positive moderator coefficient. For conservatism, a bottom peaked axial power shape was assumed for the scram reactivity model. The CEA withdrawal is initiated from the power-dependent insertion limit. The reactivity insertion rate is based on calculated CEA worths and associated uncertainties, and on the maximum withdrawal rate capability of the CEA drive system. The transient is terminated with a minimum DNBR well above 1.24 in the hot channel and with fuel temperatures well below centerline melt.

The basis for acceptance in the staff review is that the staff has reviewed and approved ABB-CE's analysis method, the input parameters are suitably conservative, and the results show that no fuel damage occurs. The calculations contain sufficient conservatism with respect to input assumptions and models to ensure that fuel damage will not result from control rod withdrawal errors. ABB-CE has met the requirements of GDC 20 and 25.

15.2.4.3 Single CEA Drop

The most limiting CEA misoperation event is the single CEA drop. If the increase in radial peaking factor is large enough, the reactor trips and there is no appreciable decrease in thermal margin. The most limiting CEA misoperation event is the single CEA drop which does not cause a trip, but results in an approach to the DNBR criterion of 1.24. The transient is initiated by the release and subsequent drop of a CEA with a resultant increase in the hot pin radial peaking factor coupled with a return to 102 percent of full power.

ABB-CE used the CESEC-III computer program to analyze the nuclear steam supply system response. The time-dependent thermal margin on DNBR was calculated using the CETOP computer program with the CE-1 critical heat flux (CHF) correlation. The sets of initial conditions (power, pressure, temperature, coolant flowrate, radial peaking factors, and axial power distribution) were chosen so that a minimum initial thermal margin was obtained. This information was then used with the maximum change in the integrated radial peaking factor. The results indicate that an increase of 14 percent in the integrated radial peak, in conjunction with the assumed values of the other initial parameters, can be tolerated without a reactor trip.

A minimum DNBR of 1.31 is reached in approximately 200 seconds. The pressure decrease beyond this point is arrested by the return to full power and a new steady state is reached. The peak linear heat generation rate during the transient remains below 21 kw/ft, thus ensuring no centerline fuel melt. The acceptance guidelines on fuel performance in the SRP are, therefore, met.

The staff reviewed the CEA misoperation events detailed in the CESSAR-DC and finds acceptable the general approach used to establish that during the most limiting events, no violations of the specified acceptable fuel design limits on DNBR, centerline fuel temperature, and RCS pressure occur. ABB-CE has complied with the requirements of GDC 25, based on conformance with the acceptance criteria in the SRP.

15.2.4.4 Startup of an Inactive Reactor Coolant Pump

ABB-CE has provided a qualitative analysis for the startup of an inactive reactor coolant pump (SIRCP) event in CESSAR-DC Section 15.4.4. This event is not a limiting transient, with respect to RCS pressure and fuel performance criteria, among the events in the same group category which will result in an increase in core reactivity. The event was evaluated during Modes 3 through 6 (hot standby, hot shutdown, cold shutdown, and refueling, respectively), since plant operation with less than all four reactor coolant pumps is permitted only during those modes of operation. For Modes 3 and 4, the primary safety valves, main steam safety valves, and the reactor protection system are designed to maintain the RCS below 110 percent of design pressure. During Modes 5 and 6, when the shutdown cooling system is aligned, overpressure protection is provided by the shutdown cooling system (SCS) relief valves. For Modes 3 and 4, the heat imbalance due to the SIRCP is less limiting than that caused by the CEA withdrawal event.

Thus, the maximum RCS pressure is maintained below 110 percent of design pressure. In Modes 5 and 6, the capacity of the SCS relief valves will prevent the RCS pressure following the SIRCP from exceeding the pressure/temperature limits for these modes.

Regarding the approach to fuel design limits for the SIRCP, the minimum DNBR in the hot channel will increase as the transient progresses. Therefore, no fuel damage is expected.

On these bases, ABB-CE's analyses conform with the acceptance criteria of SRP Sections 15.4.4 and 15.4.5 and, therefore, are acceptable.

15.2.4.5 Inadvertent Boron Dilution

In SRP Section 15.4.6, the staff requires that at least 15 minutes is available from the time the operator is made aware of an unplanned boron dilution event to the time a total loss of shutdown margin (criticality) occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. A warning time of 30 minutes is required during refueling. The staff also requires that redundant control room alarms be available to alert the operating staff to boron dilution events in all modes of operation.

In response to a staff request, ABB-CE indicated that the following pre-trip alarms are available for operational Modes 1 and 2: a high power or, under certain conditions, a high pressurizer pressure pre-trip alarm in Mode 1 or a high logarithmic power pre-trip alarm in Mode 2. Furthermore, a high RCS temperature signal may also alarm before a reactor trip. In operational Modes 3 through 6, either a high neutron flux alarm or a reactor makeup water flow alarm will alert the operator to an unplanned boron dilution event. In Modes 3, 4, and 5, with the RCS full and at least one reactor coolant pump (RCP) operating, a high neutron flux (boron dilution) alarm will indicate a boron dilution event. In Modes 4 and 5, with the RCS full and all RCPs idle, or for Mode 5, with the RCS partially drained for system maintenance, deboration is prohibited. Therefore, the reactor makeup water flow alarm will indicate any boron dilution event. In Mode 6, the boron concentration is at least 4000 ppm before entering this mode and deboration is prohibited. Therefore, the reactor makeup water flow alarm will indicate a boron dilution event.

In the DSER, the staff asked ABB-CE to submit an analysis to show that an operator has 30 minutes to act if a boron dilution alarm is used in place of the reactor makeup water flow alarm. This was designated as DSER Open Item 15.2.4.5-1. Depending on the mode of operation, a number of alarms are available to alert an operator to a boron dilution event. In addition to the alarms just mentioned, there are also sampling and boronometer indications that will provide information in case of a boron dilution event.

However, to address EPRI's advice to reduce the number of alarms presented to operators, ABB-CE performed a confirmatory analysis to show that an operator has 30 minutes to act if a neutron flux alarm is used in place of the reactor makeup water flow alarm in modes other than Mode 6. The results of this analysis were presented in Amendment N to the CESSAR-DC, and the staff finds them acceptable. On this basis, DSER Open Item 15.2.4.5-1 is resolved.

Analysis of deboration events initiated during each of the plant operational modes has shown that Mode 5 in the drained-down configuration gives the shortest available time for detecting and terminating the event. The minimum possible time interval to dilute from 5.75-percent delta-rho (Δ-ρ) subcritical (minimum shutdown margin in Mode 5) to criticality is 67 minutes. ABB-CE has shown that the redundant, qualified neutron flux alarm will alert the operator to initiation of the event in sufficient time to ensure detection of the most limiting boron dilution event at least 30 minutes preceding possible criticality. The operator can then terminate the event before shutdown margin is completely lost.

Therefore, an inadvertent deboration event will result in acceptable consequences. Sufficient time is available to meet the SRP requirements and enable the operator to detect and terminate the event before shutdown margin is lost.

15.2.4.6 Inadvertent Loading of Fuel Assembly Into the Improper Position

Most of the fuel assembly misloadings that can be postulated are easily detectable both during the CEA symmetry checks and during power range operation. However, a small number may be undetectable early in cycle. The worst case would be the interchange of a shimmed assembly with an unshimmed assembly at the center of the core. This case, although not detectable at the beginning of cycle (BOC), would cause local power peaking as the shims burn, indicating to the operator the possibility of a fuel misloading.

In addition, technical specifications require that the planar radial peaking factor be measured at least once every 31 effective full-power days (EFPD) and that the measured value be less than or equal to the value used in the COLSS and in the CPC. Therefore, even if the increase in radial peak is not large enough to indicate the possibility of a misloading, the measured radial peak would be used in the COLSS and CPC. This would reduce the operating band to compensate for any reduction in thermal margin caused by misloadings.

15.2.5 Increase in Reactor Coolant System Inventory

In CESSAR-DC Section 15.5, ABB-CE includes the analytical results for the following cases resulting in an increase in RCS inventory:

(1) inadvertent operation of the ECCS

(2) chemical and volume control system (CVCS) — pressurizer level control system (PLCS) malfunction with LOOP

ABB-CE identified that Case 2 is more limiting than Case 1 (the inadvertent operation of the ECCS) because for Case 1 the shutoff head of the high-pressure safety injection (SI) pump is less than the RCS pressure during full power, resulting in no injection of fluid into the RCS. For reactor shutdown conditions with the RCS pressure below SI pump shutoff head pressure, SI flow will increase RCS inventory and pressure. Under these conditions the shutdown cooling system (SCS) relief valves will be used for mitigation of the consequences of the event. The staff evaluation of the SCS relief valves design is reported in Section 5.2.2 of this chapter.

ABB-CE evaluated possible single failures listed in CESSAR-DC Table 15.0-4 and concluded that none of the single failures will result in a lower DNBR than that predicted for the PLCS malfunction with a loss-of-offsite-power coincident with turbine trip. For the peak system pressure consideration, ABB-CE identified failure of the proportional heaters to turn off as the limiting single failure.

For the limiting event of RCS inventory increase, the analysis assumed that when the pressurizer level controller fails low or the level setpoint generated by the reactor regulating system fails high, a low-level signal can be transmitted to the controller, which will operate the charging pump at the maximum rate [568 L/minute (150 gpm)] and close the letdown control valve to its minimum opening [114 L/minute (30 gpm)] to maximize the mass addition rate to the RCS. The loss of offsite power was assumed to be coincident with a turbine trip.

The increase in RCS inventory results in a pressurizer pressure increase to the high-pressure trip setpoint and trips the reactor. The calculated maximum RCS pressure during the transient is 1.85×10^4 kPa (2,682 psia). The 1,232 kg (2,713 lb(mass)) of steam calculated to be discharged through the pressurizer safety valves are contained within the in-containment refueling water storage tank with no releases to the atmosphere. The main steam safety valves (MSSVs) discharge 60,900 kg (134,000 lb(mass)) of steam to the atmosphere prior to 1,800 seconds. At 1,800 seconds, the operator stabilizes the plant and initiates plant cooldown using the atmosphere. Since this transient causes an increase in RCS pressure in response to an increase in reactor coolant inventory, the DNBR increases from the initial conditions. Therefore, the acceptance criteria are complied with because the calculated peak RCS pressure is within 110 percent of design pressure and no fuel failures are calculated. Therefore, the staff concludes that the analysis is acceptable.

15.3 Accident Analyses

ABB-CE analyzed accidents that are not expected to occur during the life of the plant. For accident conditions, the reactor coolant pressure should stay below the applicable ASME Code limits. The core geometry should be maintained so that there is no loss of core cooling capability.

Radiological consequences are discussed in detail in Section 15.4 of this chapter.

15.3.1 Steamline Break Analysis

In CESSAR-DC Section 15.1.5, ABB-CE analyzed six cases of steamline break (SLB) events: four to maximize potential post-trip return to power and two to maximize potential for degradation in fuel performance (i.e., with respect to DNBR) and offsite dose. To maximize the post-trip return to power, the following cases were analyzed:

(1) an SLB inside the containment at full power with concurrent LOOP in combination with a single failure and a stuck CEA

(2) an SLB inside the containment at full power in combination with a single failure and a stuck CEA

(3) an SLB inside the containment at zero power with concurrent LOOP in combination with a single failure and a stuck CEA

(4) an SLB inside the containment at zero power in combination with a single failure and a stuck CEA

To maximize the potential for fuel degradation and dose at the site exclusion area boundary, the following two cases were also analyzed:

(5) an SLB outside the containment at full power in combination with a LOOP and a single failure, a stuck CEA, and TS SG leakage

(6) an SLB outside the containment upstream of the MSIV at zero power with concurrent LOOP in combination with a single failure, iodine spike, TS SG leakage, and a stuck CEA

The largest possible SLB size is the double-ended rupture of a steamline upstream of the main steam isolation valve (MSIV). In the System 80+ design, an integral flow restrictor exists in each SG outlet nozzle. The largest effective steam blowdown area for a steamline, which is limited by the flow restrictor throat area, is approximately 30 percent of the steamline cross-section area, or 0.119 m^2 (1.28 ft^2).

Initial Conditions and Analytical Assumptions

Steamline breaks result in a rapid decrease in reactor coolant temperature and SG pressure. The RCS temperature decrease causes positive moderator reactivity feedback. The SG pressure decrease initiates a reactor trip when low pressure in the SG system produces a trip signal. From an overcooling consideration, the limiting SLB is an event with the double-ended rupture of a steamline upstream of the MSIV. The flow restrictor limits the blowdown area to 0.119 m^2 (1.28 ft^2). ABB-CE assessed ranges of parameters listed in CESSAR-DC Table 15.0-3 in establishing the worst initial plant conditions for post-trip return to power during SLBs (Cases 1 through 4). Among the worst initial conditions are the maximum core power, most positive axial shape index (ASI), maximum core inlet coolant temperature, minimum core flow rate, maximum RCS pressure, maximum pressurizer water level, and maximum water level in the SG.

For the consideration of the worst degradation in fuel performance (Cases 5 and 6), ABB-CE chose initial conditions for RCS pressure, temperature, core flow, and power to meet the following criteria: (1) to make the initial state near a power operating limit for the values of ASI and radial peaking factors used and (2) to maximize the decrease in DNBR.

ABB-CE used a three-dimensional peaking factor of 150 for post-trip return-to-power DNBR calculations. In the DSER, the staff asked ABB-CE to submit a discussion of the calculational method that determines the peaking factor of 150 and to justify the conservatism of the value used for the SLB analysis. This was designated as DSER Open Item 15.3.1-1.

By letter dated November 24, 1992, and by CESSAR-DC Section 15.1.5.3 (Amendment N), ABB-CE indicated that the peaking factor of 150 used in the SLB analysis is substantially larger than the maximum peaking factor that has been calculated for any ABB-CE plant. Typical calculated values of the maximum peaking factors in existing ABB-CE plants obtained using the ROCS/MC design methodology range from 25 to 75. These are applicable to core conditions encompassing a broad range of fuel management schemes, various burnable absorbers and conditions with insertion of all control rods except for the most reactive rod stuck out at 20 °C (68 °F). The staff finds that the peaking factor of 150 is calculated for conditions representing SLB core conditions and was previously approved by NRC for use in SLB analyses for the existing ABB-CE plants. Therefore, the staff determines that use of the peaking factor of 150 is acceptable for the System 80+ SLB analysis. On this basis, DSER Open Item 15.3.1-1 is resolved.

The CEA worth (N-1 condition for all-rods insertion with the highest reactivity rod stuck out) used in the SLB analysis is 10 percent Δ-ρ for System 80+ as compared to 8.86 percent Δ-ρ used in the SLB analysis for System 80. The increase of the CEA worth reflects a change in the fuel management for the System 80+ core from out-in to low leakage. In the System 80+ low leakage fuel management scheme, fresh fuel assemblies are placed in the interior of the core rather than on the periphery. As a result, a greater percentage of the fresh fuel assemblies are covered by CEAs, and the N-1 condition results in uncovery of fewer contiguous fresh fuel assemblies than that for the System 80 out-in fuel management scheme. This results in an increase in net CEA worth. Another contributing factor to the increase in net CEA worth is the change from boron carbide (B_4C) to erbium (Er) for the burnable absorber material. Use of Er permits better control of the power distribution peaking than that with B_4C, which in turn allows a lower leakage fuel management and thus a greater net CEA worth. As shown in CESSAR-DC Table 4.3-7, the calculated CEA worth for the N-1 condition is 10.7-percent Δ-ρ. Therefore, use of 10-percent Δ-ρ is conservative for the SLB analysis.

Other assumptions used in the analyses are use of the least-negative Doppler coefficient and most-negative moderator coefficient to maximize the core reactivity feedback, core heat flux, and minimize the DNBR. The most reactive control rod was assumed to be held in the fully withdrawn position. No operator actions were assumed within 30 minutes following an SLB.

Single-Failure Effects

ABB-CE performed a parametric study to assess the limiting single failure for a postulated SLB. For SLBs with concurrent LOOP (Cases 1, 3, and 6), the limiting single failure is the failure of one of the emergency diesel generators to start and the consequent loss of two safety injection (SI) pumps following SI actuation signal (SIAS). For SLBs without LOOP (Cases 2 and 4), the worst single failure is the failure of the MSIV on one of the steamlines on the intact generator to close following the main steam isolation signal (MSIS). Consequently, steam continues to be released from the intact SG after the MSIS. For Case 5, ABB-CE indicated that there is no single failure that increases the potential for degradation in fuel cladding performance or increases the offsite dose.

Analytical Methods and Results

The computer code used in the SLB analysis is the SLB version of the CESEC-III code, which the staff previously approved for the Palo Verde SLB analysis. In order to maximize the cooldown rate, the System 80+ specific model assumes that emergency feedwater (EFW) is actuated instantaneously to both SGs at the time of reactor trip. The maximum value of EFW is assumed to be delivered to both SGs until the operator takes manual actions to isolate EFW to the ruptured SG and begins an orderly cooldown to the shutdown cooling entry conditions.

Reactor trip as a consequence of an SLB is produced by one of several available reactor trip signals including low steam generator pressure, low RCS pressure, low steam generator water level, high reactor power, low DNBR trip initiated by the CPCs, and, for inside containment breaks, high containment pressure. Following the reactor trip, the most active control rod is assumed stuck out. For an SLB with a concurrent LOOP, ABB-CE assumed that turbine stop valve closure, which terminates feedwater to both SGs, and coastdown of the RCPs occur simultaneously. The depressurization of the affected SG results in the actuation of the MSIS, which closes the MSIVs, isolating the affected SG from blowdown, and closes the main feedwater isolation valves, terminating main feedwater to both SGs. The pressurizer pressure decrease initiates an SIAS, which introduces safety injection boron, causing core reactivity to decrease. Operator action is assumed to be delayed until 30 minutes after initiation of an SLB. The plant is cooled to 177 °C and 2.28 x 10^3 kPa (350 °F and 330 psia), at which point shutdown cooling could be initiated.

The analytical results indicated that for SLBs with concurrent LOOP (Cases 1, 3, 5, and 6) the reactor trips were initiated by CPCs in response to low RCP shaft speed. With the offsite power available, the reactor trips were initiated by CPCs as a result of a high core power condition for SLBs at full-power initial conditions (Case 2), and initiated by a low SG pressure trip signal for an SLB at zero power initial conditions (Case 4).

The analytical results demonstrate that Case 2 bounds Case 1 and Cases 3 through 6 from a return-to-power consideration. The staff finds in the analytical results that the limiting Case 2 (an SLB at full power with a single failure) does not result in a return to criticality. The maximum total reactivity for Case 2 is -0.81-percent Δ-ρ, showing that the core is subcritical and that no fuel experiences DNB.

Case 5 was cited as the limiting SLB for worst radiological consequences. The staff finds in the analytical results that Case 5 (an SLB outside the containment during full-power operation with a LOOP and a single failure) results in minimum DNBR of 1.25. No fuel failure was predicted. However, for radiological calculations, 0.5 percent of the total number of fuel rods were assumed to fail.

Staff Evaluation

The staff reviewed the SLB analysis described in CESSAR-DC Section 15.1.5 and found that approved methods (the SLB version of CESEC) were used to analyze the SLB events. The plant parameters used in the SLB analysis reflect the System 80+ design. The analytical results demonstrate that the consequences of postulated SLBs comply with the requirements in GDC 27, 28, 31, and 35 regarding control rod insertability and core coolability. Therefore, the staff concludes that the SLB analysis is acceptable.

The staff discusses its evaluation of the radiological release consequences for the SLBs in Section 15.4 of this chapter.

Since no fuel failure is predicted, ABB-CE did not use the statistical convolution method in the analysis.

In the DSER, the staff noted that ABB-CE credited the non-safety-grade turbine stop and control valves in the original SLB analyses to isolate the steam blowdown from the intact SG for an SLB with an opened MSIV in the intact SG. In the EPRI URD (Section 2.3.2.1 and Table 1.2.1), EPRI specifies use of only safety-related equipment for transient and accident mitigation. This EPRI requirement is consistent with the staff position. In the DSER, the staff required that ABB-CE reanalyze the SLB events, taking credit for only safety-grade systems. All non-safety-grade components and systems called for should be assumed to be not functional. This was designated as DSER Open Item 15.3.1-2.

In response to the staff's request, ABB-CE credited only the safety-grade systems in the reanalysis. The cases affected are full-power and zero-power SLBs without LOOP assuming the single failure of an MSIV on the intact SG to close. For full-power case 2, the turbine admission and control valves were assumed failed to close. The reanalysis assumed that the flow continues from the intact SG through the full area of the SG outlet nozzle for one steamline of 0.119 m^2 (1.28 ft^2). For zero-power case 4, the turbine admission and control valves were assumed to be closed before the initiation of the event and the flow from the intact SG was assumed to be 11 percent of the design flow, representing the maximum nonisolable steam flow with an MSIV failure to close. This flow path was represented by an effective flow area for steam blowdown from the intact SG of 0.0246 m^2 (0.2663 ft^2). The staff evaluated ABB-CE's reanalysis and concludes that it is acceptable. On this basis, DSER Open Item 15.3.1-2 is resolved.

15.3.2 Feedwater-Line Break Analysis

In CESSAR-DC Section 15.2.8, ABB-CE presents the analytical results of feedwater-line break (FLB) accidents. For the FLB analysis, ABB-CE takes no credit for a high containment pressure trip should that trip signal occur before complete depletion of the inventory of the broken SG. The analysis credited a high pressurizer pressure signal and a low steam generator water level signal for the reactor trip.

ABB-CE assessed the range of values in CESSAR-DC Table 15.0-3 to establish the worst initial conditions during an FLB. For the overpressurization concern, the initial primary system pressure was adjusted within the range specified in CESSAR-DC Table 15.0-3 to achieve a coincident reactor trip signal on high pressurizer pressure and low SG water level. This assumption maximizes the primary pressurization potential of the FLB accident, by maximizing the primary system pressure at the time of the coincident reactor trip signal. For the concern of fuel failure, the initial pressure was assumed at the minimum allowable pressure of CESSAR-DC Table 15.0-3. The assumption of the lowest pressure minimizes the pressure at time of trip and minimizes the transient DNBR.

In the FLB analysis, ABB-CE assumed that a LOOP will occur following a turbine trip caused by a reactor trip, and one emergency feedwater pump will fail to start as a result of a LOOP. Also, the range of single failures specified in CESSAR-DC Table 15.0-4 was assessed in establishing the worst single failure to maximize consequences of the accident. ABB-CE determined that none the of single failures will result in a higher RCS pressure or a lower minimum DNBR predicted for the FLB accident with combination of a LOOP.

In the DSER, the staff asked ABB-CE to justify that its FLB method is conservative as it is compared with the Semiscale test data discussed in NUREG/CR-4945 (U.S. Nuclear Regulatory Commission, "Summary of Semiscale Program (1985-1986)"). If the method were nonconservative, ABB-CE was required to reanalyze the FLB event by using the model that is supported by the test data including the Semiscale data. This was designated as DSER Open Item 15.3.2-1.

Given the assumptions discussed above, ABB-CE used the previously approved CESEC-III code to analyze a spectrum of break sizes. The Henry-Fauske critical flow model was used to calculate the FLB blowdown flow assuming saturated liquid discharge before depletion of the liquid from the affected SG and saturated steam discharge afterward. The FLB blowdown models resulted in high mass flow and low energy flow from the SG, thereby minimizing the ruptured SG heat removal capacity. By letter dated December 18, 1992, and in CESSAR-DC Section 15.2.8.3-A (Amendment U), ABB-CE stated that for the FLB analysis, the heat transfer area was assumed at design value until the SG liquid inventory decreased to 225 kg (500 lbm). The heat transfer area is then decreased to zero over the time interval for inventory to decrease by 225 kg (500 lbm). The value of 225 kg (500 lbm) represents about 0.2 percent of the initial inventory. This assumption represents the decrease in heat transfer area to zero in about 0.2 second for the limiting break flow rate. The staff finds that the heat transfer model discussed above is consistent with Semiscale test for FLBs included in Section 4.3.3.1 of NUREG/CR-4945. These data indicate that the SG heat transfer capacity remains unchanged until the SG liquid inventory is nearly depleted. This is followed by a rapid reduction to zero heat transfer with little further reduction in the SG liquid inventory. Therefore, the staff concludes that the heat transfer model is acceptable for the FLB analysis. On this basis, DSER Open Item 15.3.2-1 is resolved.

ABB-CE performed the FLB analysis for the full spectrum of break sizes up to the double-ended guillotine break with an effective break area of 0.13 m^2 (1.4 ft^2). The results of the analysis show that the maximum peak RCS pressure, which is 1.92 x 10^4 kPa (2798 psia), occurs for a 0.056 m^2 (0.6 ft^2) break downstream of the check valves in the feedwater line. This peak pressure is well within 120 percent of the primary system design pressure, and conforms to the criteria of SRP Section 15.2.8, Item II.D.1, which limits the system pressurization to 120 percent of the design pressure for very low probability events. The staff considers that an FLB accident with a LOOP is a very low probability event. In response to the staff's request, ABB-CE performed an analysis (CESSAR-DC Section 15.2.8, Amendment U) for an FLB accident with the offsite power available and an assumed loss of one emergency feedwater pump as the limiting single failure. The analysis credited a low SG water level signal actuated at 33.7 percent of the wide-range SG water level. The analysis shows that the peak pressure of 1.85 x 10^4 kPa (2676 psia) is within 110 percent of the design pressure and demonstrates the compliance of SRP Section 15.2.8, Item II.D.1, which allows the system pressurized up to 110 percent of the design pressure for low probability events. The staff considers that an FLB accident with SF and available offsite power is a low probability event.

ABB-CE's DNBR calculations show that the limiting case is a 0.019 m^2 (0.2 ft^2) break with a LOOP following turbine trip, resulting in a minimum DNBR of 1.17. ABB-CE used the statistical convolution method to calculate the number of the failed rods that experienced DNB, and determined that 0.60 percent of the fuel experienced cladding failure.

Since the FLB analysis uses approved methods to show that the peak pressure meets the acceptance criteria of SRP Section 15.2.8, and the radiological releases are within the limits of 10 CFR Part 100, the staff concludes that the feedwater line break analysis is acceptable. (See Section 15.4 of this chapter for the staff's evaluation of the radiological release calculation).

15.3.3 Single Reactor Coolant Pump Shaft Seizure and Shaft Break

The RCP shaft seizure and shaft break are classified as limiting-fault events. In accordance with GDC 17, such events should be analyzed assuming a LOOP throughout the events and the worst single failure of an active component.

In CESSAR-DC Sections 15.3.3 and 15.3.4, ABB-CE analyzed the RCP shaft seizure and shaft break events with a LOOP and single-failure consideration. The analyses show that the RCP shaft-seizure event with a LOOP bounds the RCP shaft-break event with a LOOP since the RCP flow coastdown for the shaft- seizure event with a LOOP is faster, resulting in a lower minimum DNBR and more radiological release than those of the shaft-break event with a LOOP. ABB-CE submitted a detailed analysis of the bounding case of the RCP shaft seizure in CESSAR-DC Section 15.3.3.

A single RCP shaft seizure can be caused by seizure of the upper or lower thrust-journal bearings. A LOOP will cause a simultaneous loss of feedwater flow, condenser inoperability, and coastdown of all reactor coolant pumps. In the analysis, ABB-CE took no credit for restoring offsite power before initiating shutdown cooling.

For the single RCP shaft-seizure event, the reactor was tripped on low reactor coolant flow, and the resulting minimum DNBR occurred during the first 4 seconds of the event. ABB-CE evaluated the single-failure events in CESSAR-DC Table 15.0.4. ABB-CE stated that an ADV failing to close 1800 seconds after initiation of the event is the most limiting single failure. The stuck-open ADV causes excessive steam to be released to the environment from the SGs. Thus, this failure in combination with the LOOP maximizes the radiological consequences of the event.

ABB-CE analyzed the RCP shaft seizure with a LOOP using the CESEC-III code for calculating the system response; the HERMITE code for calculating reactor core neutronic parameters; the TORC code for conducting the core thermal-hydraulic analyses; and the CE-1 correlation for determining the DNBR. The calculated results showed that the maximum RCS pressure is 1.82 x 10^4 kPa

(2635 psia), which is less than 110 percent of design pressure, and the minimum DNBR is 1.09. ABB-CE used the statistical convolution method to determine the number of failed rods for the RCP-shaft seizure event with a LOOP. The results show that no more than 1.2 percent of the fuel pins would potentially fail. As discussed in Section 15.1 of this chapter, the staff approves the application of the statistical convolution method for failed-rod calculations.

Also, ABB-CE assumed that the LOOP occurs coincidently with a turbine trip. As discussed in Section 15.1 of this chapter, the staff determines that this approach is consistent with the staff's position and is, therefore, acceptable.

Since the NRC-approved methods are used to show that the peak pressure is within 110 percent of the design pressure and the limiting conditions are identified for radiological release calculations, the staff concludes that the applicant's analysis for the RCP shaft seizure with a LOOP is acceptable. The staff reviews the radiological releases appears in Section 15.4 of this chapter.

15.3.4 Control Element Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a CEA. For CEAs initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions make this accident extremely unlikely, ABB-CE analyzes the consequences of such an event.

ABB-CE reported on methods used in the analysis in CENPD-190-A; the staff has reviewed this report and accepted it. This report demonstrates that the model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The initial conditions examined in the CESSAR-DC range from zero-power to full-power with reactivity coefficients representative of beginning of cycle and end of cycle for these power level extremes. All cases resulted in a calculated radial average fuel enthalpy for the hottest fuel pellet less than the Regulatory Guide (RG) 1.77 acceptance criterion of 280 cal/gm. In addition, for the case initiated from full power, initial conditions resulted in the largest number of fuel cladding failures, approximately 6.8 percent of the fuel, and, therefore, the greatest potential for offsite dose consequences.

For a CEA ejection accident, the staff has traditionally assumed, for the purpose of calculating dose, that a fuel rod will fail if its DNBR falls below the approved DNBR limit value. ABB-CE assumes that the number of failed fuel rods equals the number of rods in DNB as calculated with the statistical convolution method described in CENPD-183. That is, since the probability of occurrence of DNB is a function of the DNBR, the statistical convolution technique involves the summation over the reactor core of the number of rods with a specific DNBR multiplied by the probability of DNB at that DNBR. Since this deviation has been appropriately justified (as discussed in Section 15.1.1 of this chapter), it is acceptable for this event analysis. On this basis, the resulting doses are well within the guidelines of 10 CFR Part 100 and, therefore, acceptable.

The staff reviewed the ejected rod worths and reactivity coefficients used in the analysis and judges them conservative. The assumptions and method of analysis used by ABB-CE are also in accordance with those recommended in RG 1.77 or comply with subsequent staff positions. Therefore, this analysis is acceptable.

15.3.5 Inadvertent Opening of a Pressurizer Safety Valve

ABB-CE categorized the inadvertent opening of a pressurizer safety valve (IOPSV) event as a limiting fault event since no power-operated relief valve (PORV) is included in the System 80+ design. In SRP Section 15.6.1, the staff identified the inadvertent opening of a PORV as a moderate-frequency event. Since the pressurizer safety valve is a spring-loaded valve and is more reliable than the PORV, it is less likely to be opened inadvertently. ABB-CE's categorization of the IOPSV event is acceptable. In CESSAR-DC Section 15.6.1, ABB-CE evaluated the IOPSV event for the System 80+ design and determined that the results of this event are bounded by that of the small-break LOCA analysis because the pressurizer safety valve size is within the range of small-break sizes. The staff agrees with this conclusion. The staff's review of the small-break LOCA analysis is discussed in Section 15.3.7 of this chapter.

15.3.6 Double-Ended Break of a Letdown Line Outside the Containment

Reactor coolant may be directly released from a break or leak outside of the containment in a letdown line, instrument line, or sample line. In CESSAR-DC Section 15.6.2, ABB-CE states that the worst event is the double-ended break of the letdown line outside the containment, upstream of the letdown line control valve (DBLOCUS). ABB-CE considers this as the worst event because this event is the largest letdown line break and results in the largest release of the reactor coolant outside the containment.

The results of ABB-CE's analysis [CESSAR-DC Figure 15.6.2-6 (Amendment N)] shows that a DBLOCUS event releases the RCS primary fluid at a rate of approximately 12.3 kg/second (27 lb(mass)/second). The maximum break flow is limited to this value by the use of the letdown line orifices inside the containment, downstream of the letdown heat exchanger. The event will actuate a number of alarms that would be noted by the reactor operator in the main control room. Within a few minutes after initiation of the event, the following alarms will be actuated to alert the operator: (1) the letdown line low-pressure alarm; (2) the high-temperature, high-humidity, and high-radiation-level alarms in the nuclear annex; and (3) the pressurizer low-level, nuclear annex sump high-level and the volume control tank low-level alarms.

ABB-CE assumed that 30 minutes after the first alarm, the operator would isolate the letdown line, thereby terminating further release of primary fluid discharged to the nuclear annex, and subsequently bring the reactor into the shutdown condition.

ABB-CE assessed the range of parameters in CESSAR-DC Table 15.0-3 in establishing the most adverse initial condition for the maximum total mass release. The worst initial conditions comprise (1) maximum core power, (2) maximum core inlet temperature; (3) low core flow, (4) maximum pressurizer pressure, and (5) high pressurizer level. ABB-CE also assumed that the CVCS charging pump flow was at the minimum design flow rate in order to maximize the letdown line discharge enthalpy and flashing and, thus, maximize the radiological consequences.

ABB-CE used the NRC-approved CESEC-III code to simulate the event and calculated the reactor coolant discharge outside the containment to use for calculating radiological release. The staff's evaluation of the radiological release calculations is discussed in Section 15.4 of this chapter. Since the blowdown rate and the rate of decrease of RCS pressure (which determines the extent of decrease in the DNBR) during this event are bounded by that of SGTR events, the minimum DNBR resulting from the event with and without a LOOP is limited by that of the SGTR event with a double-ended tube rupture (which results in a blowdown rate [(CESSAR-DC Figure 15.6.3-42A (Amendment H)] of approximately twice that for the DBLOCUS event) and, thus, does not fall below the DNBR safety limit of 1.24, assuring the fuel integrity throughout the event.

Since the assumptions used and the analyses performed for this event are acceptable, and the scenario, as described in CESSAR-DC Section 15.6.2, assures that ABB-CE considered the most severe failure of a letdown line carrying the primary coolant outside the containment, the staff concludes that the analysis is acceptable.

As stated in the DSER, ABB-CE was required to submit the technical basis justifying that the valves in the letdown line, instrumentation line, or sample line have been qualified to be isolated upon demand during piping break conditions. This was designated as DSER Open Item 15.3.5-1.

In response, ABB-CE stated that the isolation valves of the concern will be qualified to close on demand. For example, as stated in CESSAR-DC Section 9.3.4.2.3 (Amendment U), three isolation valves in the letdown line are manually closed following detection of a letdown line break. Valve closure is ensured by specifying valve operators sized to close the valves under the worst-case differential pressures expected during applicable design-basis events. Valve operability under these conditions is assured via the operability assurance program (OAP) for pneumatically operated valves, as discussed in CESSAR-DC Section 3.9.3.2.1.1. These valves are tested in accordance with the requirements specified in CESSAR-DC Table 3.9-15, "Inservice Testing (IST) for Safety Related Valves." Since ABB-CE determines the size of valve operators on the basis of the worst differential pressure during the design-basis events and uses the OAP and IST to ensure the operability of the isolation valves, the staff concludes that the isolation valves are adequately qualified for closure on demand. On this basis, DSER Open Item 15.3.5-1 is resolved.

15.3.7 Loss-of-Coolant Accident

In CESSAR-DC Section 6.3.3, ABB-CE presents results of the LOCA analysis. ABB-CE performed the LOCA analysis using the following NRC-approved evaluation models, as discussed in Section 15.1 of this chapter: CEFLASH-4A large-break LOCA (LBLOCA) and CEFLASH-4AS small-break LOCA (SBLOCA) for the system response during the blowdown phases, STRIKIN-II for calculating the hot rod cladding temperature, HCROSS and PARCH for calculating the steam cooling heat transfer coefficient, and COMZIRC for determining the core-wide cladding oxidation.

The acceptance criteria for LOCAs given in 10 CFR 50.46 are:

(1) <u>Peak Cladding Temperature</u> — "The calculated maximum fuel element cladding temperature shall not exceed 1,204 °C (2,200 °F)."

(2) <u>Maximum Cladding Oxidation</u> — "The calculated total oxidation of the cladding shall nowhere exceed

0.17 times the total cladding thickness before oxidation."

(3) Maximum Hydrogen Generation — "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

(4) Coolable Geometry — "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

(5) Long Term Cooling — "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

Various break size LOCAs were analyzed and the results for large-break LOCAs (break sizes ranging from 0.046 m^2 (0.5 ft^2) in cross-section area to the double-ended cold-leg guillotine) are given in CESSAR-DC Section 6.3.3.2. The results of small-break LOCA analyses (break sizes less than 0.046 m^2 (0.5 ft^2) are documented in CESSAR-DC Section 6.3.3.3.

Because the System 80+ emergency core cooling (ECC) system has direct ECC injection into the reactor vessel downcomer, the staff reviewed the applicability of the existing ABB-CE LOCA evaluation model (EM), which was developed for plants with cold-leg injection. The LOCA EM assumes complete ECC bypass up to the end of bypass; therefore, it complies with the Appendix K requirement with regard to the blowdown phase. The staff considered whether the EM properly addresses the potential ECC bypass as the ECC is injected into the downcomer during the reflood phase.

ABB-CE's EM does not explicitly model the ECC bypass/entrainment phenomenon during the reflood phase. All safety injection other than that discounted because of the single-failure consideration is assumed to fall into the downcomer. Therefore, the downcomer liquid level is at the bottom of the cold leg, above which the ECC water spills over the broken cold leg into the containment. The staff performed an evaluation and concluded that, if the ECC bypass/entrainment phenomena were considered, there will be a small downcomer level reduction (< 1 ft) that could slightly affect the calculated reflood rate and peak cladding temperature. However, because there is large conservatism built into the Appendix K requirements, such as use of 1.2 times the 1971 ANS decay heat model and the Baker-Just metal-water reaction model, that provide a large margin to account for uncertainties associated with simplified models or neglected phenomena in the EM, the staff believes that the small uncertainty associated with neglecting the ECC bypass phenomenon is easily compensated by the built-in margin. As Appendix K is silent regarding ECC bypass during reflood period, the staff concludes that no non-compliance with Appendix K exists in ABB-CE's EM.

Large-Break LOCA (LBLOCA)

For the LBLOCA analysis, offsite power was assumed to be lost simultaneously with the LOCA. ABB-CE determined that the LBLOCA with the maximum safety injection (as a result of no-single-failure assessment) is the limiting case. The maximum safety injection maximizes the safety injection spilling to the containment and minimizes the containment pressure. This, in turn, minimizes the core flooding rate and maximizes the peak cladding temperature. In the analysis, the maximum safety injection includes flow from all four SITs assuming the maximum initial liquid inventory and the maximum flow rate from all four safety injection pumps. In the DSER, the staff asked ABB-CE to perform a sensitivity study and demonstrate that the LBLOCA with the maximum SI flow is the limiting case for the System 80+ design. This was designated as DSER Open Item 15.3.6-1.

In response to the staff's request, ABB-CE submitted material dated November 24, 1992, and CESSAR-DC Appendix 6A (Amendment S) documenting the results of a sensitivity study to demonstrate that the assumption of a maximum SI flow will result in a worst-case LOCA. In the sensitivity study, ABB-CE analyzed three LOCA cases: (1) a loss of two SI pumps due to a diesel generator failure, (2) a loss of one SI pump, and (3) a maximum SI flow with four SI pumps available. For each case, ABB-CE used the staff's previously approved COMPERC-II code for the reflood and refill calculation. The study was performed for the double-ended-discharge cold-leg guillotine break with a discharge coefficient of 1.0, which was identified as the limiting break of the large-break spectrum for the System 80+. The results have demonstrated that case 3 with a maximum SI flow is the limiting LOCA case, since this case results in the lowest reflood rate which will cause the highest peak cladding temperature. On this basis, DSER Open Item 15.3.6-1 is, resolved.

Originally, ABB-CE analyzed nine LBLOCAs at the power level of 3,876 MWt. The LOCAs include slot and guillotine breaks, ranging in size from 0.046 m^2 (0.5 ft^2) to a full double-ended break area such as the reactor coolant pump suction and discharge leg. The analysis showed that the worst case is the double-ended-discharge cold-leg guillotine (DEDCLG) break. ABB-CE reanalyzed the limiting case (the DEDCLG break) at 3,992 MWt. The reanalysis shows a peak cladding temperature of 1,196 °C (2,185 °F), maximum cladding oxidation of 8.32 percent of the total cladding thickness, and metal-water reaction of less than 0.843 percent of the total amount of metal in the core. The results are within the acceptance criteria of 10 CFR 50.46 discussed above.

Small-Break LOCA (SBLOCA)

For the SBLOCA analysis, ABB-CE assumed that a LOOP occurred simultaneously with a reactor trip. The worst single failure identified is failure of one of two emergency diesels generators to start, which results in only two of four safety injection pumps are available to function. The LOCA with a LOOP upon a reactor trip and the worst single failure will minimize the safety injection available to cool the core and will result in a maximum peak cladding temperature. In the analysis, the following injection flows were credited for the SBLOCA analysis:

- For a break in the pump discharge leg, the safety injection credited was full flow from two SI pumps and four safety injection tanks (SITs).
- For a break in a direct vessel injection (DVI) line, the SI flow credited was full flow from one SI pump and three SITs. The flow from the remaining active SI pump and one SIT was assumed to spill out the break.

ABB-CE analyzed 11 SBLOCAs. Three DVI line breaks were analyzed at a core power level of 3,992 MWt (102 percent of nominal). The DVI line was determined to be the limiting break location based on an eight-break analysis performed at 3,876 MWt. Among the eight breaks, four breaks, ranging in size from 0.046 m^2 to 0.0046 m^2 (0.5 ft^2 to 0.05 ft^2), were postulated to occur in the pump discharge leg. The 0.046 m^2 (0.5 ft^2) break was also analyzed for the large-break spectrum and was defined as the transition break size. Four breaks were postulated to occur in DVI lines, ranging in size from 0.037 m^2 (0.4 ft^2) (full cross-sectional area of a DVI line) to 0.0018 m^2 (0.02 ft^2). The sizes of the three DVI line breaks analyzed at 3,992 MWt were 0.009 m^2 (0.1 ft^2), 0.007 m^2 (0.08 ft^2), and 0.0045 m^2 (0.05 ft^2). For DVI line breaks larger than 0.009 m^2 (0.1 ft^2), the SITs will operate sooner and restore water level more quickly than for the 0.009 m^2 (0.1 ft^2) break. The resulting peak cladding temperature would be less than that calculated for the 0.009 m^2 (0.1 ft^2) break. Therefore, ABB-CE did not reanalyze DVI breaks larger than 0.009 m^2 (0.1 ft^2) at 3,992 MWt. The reanalysis shows that the worst small break is a 0.009 m^2 (0.1 ft^2) DVI break which resulted in the highest cladding temperature of 734 °C (1,354 °F), maximum cladding oxidation of 0.12 percent of the total cladding thickness, and metal-water reaction of less than 0.016 percent of the total amount of metal in the core.

NRC-approved methods were used to analyze small- and large-break LOCAs; peak cladding temperatures are less than 1,204 °C (2,200 °F); metal-water reaction is within 17 percent of the total cladding thickness; and cladding oxidation is within 1 percent of the metal in the cladding cylinders surrounding the fuel. Since the results of the analysis do not exceed the acceptance criteria imposed in 10 CFR 50.46 for the LOCA analysis, the staff concludes that the SBLOCA analysis is acceptable. The evaluation for the post-LOCA long-term cooling appears in Section 15.3.8 of this chapter.

Boron Dilution During SBLOCAs

Experimental evidence and recent analysis show that an inherent mechanism for boron dilution in the PWR RCP loop seals could exist for events (including SBLOCAs) that involve heat removal by reflux cooling. The deborated water in the RCP loop seals could be transported to the core through natural circulation processes or startup of RCPs. Injection of the deborated water into the core would be a significant reactivity addition that could possibly damage the core. The staff asked ABB-CE to address the applicability of this boron dilution event to the System 80+ design and to resolve the issue.

In response to the staff's request, ABB-CE submitted the results of their evaluation of the potential for RCS boron dilution during an SBLOCA. Basically, the postulated SBLOCA scenario results in the accumulation of deborated water in each of the RCS cold-leg loop seals. The mechanism for accumulating deborated water in the loop seals is caused by steam condensation (reflux cooling) following drainage from steam generator (SG) tubes. During reflux cooling, the condensate on the cold-leg side of the SG tubes drains into the loop seals. The staff was concerned that in this configuration, the introduction of deborated water in the core would have deleterious effects on maintaining subcriticality. ABB-CE stated that low boron concentration in the System 80+ loop seals may occur for small break sizes between 2.54 and 7.62 cm (1 and 3 in.) in diameter.

A bounding analysis was performed without crediting any of the mixing of borated and unborated water which is expected to occur in the RCS. Instead, the condensate was assumed to enter the core as an unlimited size slug of pure water moving at a natural circulation flow rate consistent with that of a small break at the time of RCS refill using ECCS injection.

A core physics analysis was performed to determine the reactivity, the power peaking, the power transient and the minimum critical boron concentration required to avoid recriticality at beginning of the cycle with all rods inserted as the unborated slug passed through the core. An RCS thermal-hydraulics analysis also was performed to determine the change in pressure and in natural circulation flow rate as energy from the core entered the coolant.

The analysis indicates that the core returned to a critical condition when the unborated slug progressed partly through the core. As the slug progressed further into the core, the resultant neutron power function experienced a very brief spike, which was terminated by Doppler feedback in the fuel. The power then dropped further as coolant heatup resulted in moderator density reactivity feedback. The power underwent several oscillations of diminishing amplitude and finally settled at a level that was a small fraction of full power. The analysis indicated that boron concentration required to avoid a return to criticality at beginning of the cycle with all rods inserted, depends on the temperature of the coolant. An average boron concentration of about 550 ppm is required to avoid a return to criticality at 149 °C (300 °F), but only 200 ppm is required at 260 °C (500 °F).

ABB-CE concluded that if the analysis had accounted for borated water entering the core behind the slug, then the power would have rapidly decreased to zero. The staff finds this acceptable because during RCS refill, the SG tubes will fill through the hot leg, and the borated ECCS water (4400 ppm) would flow and mix with the deborated loop seal before entering the reactor vessel. Once in the vessel, the staff believes that additional mixing would occur through the downcomer.

For natural circulation conditions, ABB-CE calculated that the time required for the condensate in the loop seals to pass through the core is approximately 3.3 minutes and it will take two to three times longer for the condensate to pass through the RCS. ABB-CE also postulated that the consequences for restarting the RCP wouldn't be of concern if procedural restrictions delayed the restart process for at least 20 minutes under natural circulation conditions.

The staff agrees that a 20-minute delay is a conservative time limit to permit the condensate to pass through the RCS at the natural circulation flow rate (approximately 2 percent to 3 percent of total flow) and mix with the highly borated coolant in the RCS. However, the staff is concerned that the operator could err in determining that natural circulation is established, and for how long it is established. Because of the potentially serious consequences of an operator prematurely restarting an RCP (assuming the presence of an unborated slug), the staff believes that procedural controls alone may not be adequate. ABB-CE must, therefore, demonstrate that the event is incredible; the consequences are not serious; or provide additional protective measures.

SBLOCA Deboration Events With Restarting an RCP

In response to the staff's concerns, ABB-CE submitted their analysis and changes to Emergency Operations Guidelines (EOGs) described in CESSAR-DC, Appendix 6C to support their position that the SBLOCA deboration event with restart of a reactor coolant pump (RCP) is unlikely to occur. Even if RCP restart occurs under this condition, the consequences are not serious. The staff has reviewed CESSAR-DC, Appendix 6C and proposed EOG changes and provides the following evaluation.

Background

The ABB-CE system response analysis shows the potential for an SBLOCA deboration event for SBLOCA break sizes ranging from 2.54 to 7.62 cm (1 to 3 in.) in diameter. These sizes of break are small enough that the break flow is not sufficient to remove all of the decay heat. The secondary side will be relied on for the decay heat removal. Also, these break sizes are large enough so that the break flow is greater than the safety injection flow, thus reducing the reactor coolant system (RCS) water level below the bottom of the cold leg. At this water level, the steam generated in the reactor core will be transferred to the tube side of the steam generator (SG). When RCS cooling by the secondary side is initiated, the reflux/condensation process begins.

Formation of the condensate will result in some of the condensate flowing back via the hot legs to the reactor vessel, counter-current to the steam flow. The rest of the condensate will flow into the loop seals (RCS suction pipes) and collects there until the RCS refills and natural circulation is regained. If the operator inadvertently turns on the RCPs, the unborated (or low borated water) in the loop seals could be transported into the core rapidly, and cause a rapid reactivity transient.

ABB-CE performed a probabilistic risk assessment (PRA) of the SBLOCA deboration event for System 80+. Important factors in the analysis included: the likelihood of an SBLOCA, the amount of boron mixing in the cold-leg piping during the refill phase of the event, reestablishment of natural circulation in the primary system, and the likelihood of an operator restarting an RCP prior to the establishment of natural circulation.

Small break LOCA frequency is typically estimated to be in the range of 10^{-2} to 10^{-3} per reactor year, and is therefore a potentially significant initiator. Mixing in the cold-leg and the loop seal may occur during the refill phase as a result of highly borated water from the SI pumps flowing back through the RCP into the loop seal. Low rates of natural circulation will resume once the lower tubes of the steam generator are filled. The natural circulation rate (and mixing process) will increase until the system is refilled. ABB-CE's analysis indicates that when all steam generator tubes have filled, the natural circulation rates result in a well mixed primary system in about 20 minutes. ABB-CE also analyzed the reactivity effect of unborated water entering the core at the natural circulation flow rate. For the postulated case of a slug of pure (i.e., unborated) water entering the core in this manner, the resultant neutron power function exhibited a brief spike, which was terminated by Doppler feedback in the fuel. The power then dropped further as coolant heatup resulted in moderator density reactivity feedback. Since the result of the natural circulation scenario is insignificant for System 80+, the staff's concern was directed to the likelihood that an operator would start an RCP during the period after the loop seal is potentially filled with condensate but before natural circulation had been reestablished.

In response to the staff's concern, ABB-CE has changed to the System 80+ EOGs to ensure that the operator will not inadvertently turn on the RCPs during an SBLOCA event. The EOGs changes are as follows:

(1) The RCP Restart Strategy

The RCP restart strategy mainly involves five steps. Since maintenance of natural circulation (NC) prior to restart of the RCP will help operators avoid unacceptable core conditions from occurring during an SBLOCA event, the RCP restart steps were modified in the following priority in order to emphasize the importance of maintenance of NC before turning on an RCP:

(a) Verify adequate single-phase NC

(b) If single-phase NC cannot be established, verify adequate two-phase NC

(c) Determine if RCP restart is needed and desired

(d) Verify that all RCP restart criteria are met

(e) Restart RCPs.

These modifications are reflected in the LOCA recovery guideline, success paths HR-2, HR-3 and PC-5 of the Functional Recovery Guideline.

(2) RCP Restart Desirability and Criteria

Step (1)(c) above provides guidance for the RCP restart desirability and Step (1)(d) provides acceptance criteria for RCP restart. To further ensure that the operator will not inadvertently restart the RCP prior to establishment of NC, two modifications were made to each of these steps: one modification requires the operator to obtain concurrence from the technical support center (TSC) on the RCP restart; the other requires the operator and the TSC to consider the length of time that the plant had been in NC when evaluating the desirability of RCP restart.

(3) Supplementary Information Item

It is important for the operator to consider whether or not deborated water could build up in the suction leg of the RCP prior to RCP restart. ABB-CE has added supplementary information to the LOCA recovery guideline that cautions the operator about this possibility prior to RCP restart. In addition, ABB-CE has specified that the supplementary information should become a caution in the plant specific procedures. Specifically, this caution is intended to be placed prior to the step for RCP restart desirability determination (Step (1)(c) above) in the plant specific procedures.

(4) Modifications to the EOG Bases

The Bases section was modified to match the new step order and to contain the bases explanations for the new steps.

The staff has reviewed these modifications to the EOGs. Since the modified EOGs require the operator to take many steps to restart an RCP, including checking with the TSC to confirm that natural circulation has commenced, thus assuring adequate shutdown margin, the staff concludes that the modified EOGs provide a reasonable assurance that the operator will not inadvertently restart the RCP during an SBLOCA event. Therefore, the EOG changes are acceptable.

Boron Mixing Analysis

In assessing the need for modifications beyond the EOG revisions discussed above, the staff considered that the operator could err in determining that natural circulation is established and for how long it is established. ABB-CE was asked to assess the efficacy of boron mixing in the RCS assuming the presence of a large unborated slug and an operator action to start the RCP in the same loop.

As initial conditions for the analysis, ABB-CE assumed that the loop seal and the cold leg volume (below centerline) is filled with pure (unborated) water. This is a volume of 7.42 m^3 (262 ft^3). The computational fluid dynamics (CFD) code (FLUENT) was then used to assess the mixing in the downcomer and the lower plenum of the core following the startup of one reactor coolant pump (RCP). FLUENT is a thermal hydraulic code widely used in the industry to model fluid flow in pipes. It has two dimensional and three dimensional calculational capability for steady or transient flow calculations, compressible or incompressible flow modeling, and can track chemical species distributions in the fluent. Although the staff did not review the details of the code, the staff did review the various required inputs and the applicability of the code to the boron dilution problem.

ABB-CE included the following conservatisms in its analyses:

(1) No credit was taken for mixing in the RCP discharge pipe.

(2 No credit was taken for flow entrainment in the reactor vessel.

(3) No credit was taken for mixing at the reactor vessel inlet nozzle as the slug of water hits the reactor vessel plenum wall.

(4) Mixing in the reactor vessel lower head and lower support structure was assumed. This mixing was underestimated by the use of a simplified flow path which ignored the tortuous path the fluid must take in the lower head and lower support structure area.

The staff concludes that the ABB-CE analysis used conservative assumptions to assess boron mixing effects in the RCS for the SBLOCA deboration event.

The output of the FLUENT code includes boron concentration as a function of space and time. The results of the ABB-CE analysis show that the boron concentration of the water entering the core region is heavily dependent upon the initial slug volume. Doubling the initial slug volume (from 7.42 m^3 (262 ft^3) to 15.35 m^3 (542 ft^3)) reduced the minimum boron concentration by approximately 35 percent. A slug size of 15.35 m^3 (524 ft^3) reduced the calculated boron concentration in the core region to 1,350 ppm.

Critical boron concentration is the boron concentration above which criticality will not occur. The critical boron concentrations are also a function of temperature. Analysis by the ABB-CE shows that, at BOC and all rods in (ARI), at 260 °C (500 °F), the critical boron concentration is 200 ppm. At 149 °C (300 °F), the critical boron concentration is 550 ppm. These values are well below the 1,350 ppm obtained as assuming the initial presence of a large slug of 15.35 m^3 (524 ft^3). In addition, ABB-CE shows by neutronic analysis that the rapid reactivity transient may cause recriticality only during the first third of the fuel cycle since beyond this cycle time no boron is required to maintain the core subcritical for post-LOCA conditions with all control rods inserted. Therefore, the staff concludes that ABB-CE's analysis provides a reasonable assurance that sufficient boron mixing can be expected to prevent core recriticality from occurring during an SBLOCA deboration event in the unlikely event that an operator restarts an RCP before natural circulation is fully established.

<u>Conclusion for SBLOCA Deboration Events With Restarting an RCP</u>

The staff has reviewed the analysis and EOGs changes described in CESSAR-DC, Appendix 6C. As a result, the staff concludes that there is a reasonable assurance that the postulated deboration transient during an SBLOCA with an RCP restart poses no undue threat to the public health and safety.

15.3.8 Post-LOCA Long-Term Cooling

Long-term cooling (LTC) initiates when the core is quenched after a LOCA and terminates when the plant is secured. The objectives of LTC are to maintain the core at a safe temperature level and to avoid the precipitation of boric acid in the core region. In CESSAR-DC Section 6.3.3.4, ABB-CE describes the LTC methods (CENPD-254-A) for the System 80+ design.

The System 80+ design uses two different methods for LTC, depending on the break size. If the break size is sufficiently small, the shutdown cooling system (SCS) is used. For LBLOCAs, simultaneous hot-leg and direct vessel injection (DVI) are used to maintain core cooling and avoid boric acid precipitation.

The LTC operation requires the operator to initiate cooldown within 1 hour following a LOCA by releasing the steam through the turbine bypass system (if ac power is available) or through the atmospheric dump valves (if ac power is unavailable). Between 2 and 3 hours following a LOCA, the operator is required to open hot leg injection valves in charging piping of SI pumps 3 and 4, and to close the corresponding DVI flow-path valves for hot-leg injection. The DVI nozzle flow paths of SI pumps 1 and 2 are opened. This configuration with SI pumps 3 and 4 injection to the hot legs, and SI pumps 1 and 2 injection to respective DVI nozzles, provides simultaneous hot-leg injection and DVI for LTC.

Between 8 and 9 hours after the LOCA, if the RCS pressure exceeds 3.1×10^3 kPa (450 psia) and the RCS is filled with water, the operator is required to cool the plant down to the shutdown cooling conditions by using the steam generators and the pressurizer auxiliary spray. The analysis for the System 80+ design uses the criterion of RCS pressure greater than 3.1×10^3 kPa (450 psia) at 8 to 9 hours after a LOCA to distinguish the LOCA as a small-break and initiate the SCS for LTC. The SIS is designed so that one of the hot-leg injection systems and one of the DVI systems will remain functional during the worst single failure, which is identified as failure of one of two emergency diesels to start.

ABB-CE used the approved methods in CENPD-254-P-A to perform the LTC analysis demonstrating adequacy of the LTC operation strategy. For the large break, offsite power was assumed to be lost during the accident. The identified worst single failure is the failure of one of the diesel generators to start, resulting in only two SI pumps and one emergency feedwater train being available for LTC. The LTC analysis assumed one SI pump injection to spill at the break for the DVI line break and only credited one SI pump for LTC. One atmospheric dump valve on each steam generator was used to cool down the RCS. The cooldown was assumed to begin 1 hour after a LOCA.

The results of the analysis for the double-ended cold-leg break, which was identified as the worst case in terms of long-term boric acid accumulation in the inner vessel, show that the boron concentration in the core remains below the boric acid precipitation limit during post-LOCA conditions. Thus, the analysis shows that there is no threat to long-term cooling due to blockage caused by the boric acid precipitation.

The LTC analysis for the small break (size less than 0.003 m^2 or 0.03 ft^2) also demonstrates that ABB-CE will be able to use the SCS for the long-term cooling for a small-break LOCA. During the cooldown, sufficient emergency feedwater is available to cool the plant down to the shutdown cooling entry conditions, and the SI flow will refill the RCS to ensure that proper suction is available for entering shutdown cooling.

By using previously approved methods to demonstrate an adequate margin available for the post-LOCA LTC, ABB-CE complied with the long-term core cooling acceptance criteria of 10 CFR 50.46. However, the original analysis credited the auxiliary pressurizer spray, which is a non-safety-grade system, for the RCS cooldown and assumed the SCS entry conditions to be 4.19×10^3 kPa and 204 °C (608 psia and 400 °F). The indicated entry conditions for the System 80+ SCS are 3.1×10^3 kPa and 177 °C (450 psia and 350 °F). ABB-CE was required to reanalyze the post-LOCA LTC using only safety-grade systems and using the SCS for LTC based on the design initiation temperature and pressure. This was designated as part of DSER Open Item 15.3.1-2.

In response to the staff's request, ABB-CE's reanalysis for the LOCA long-term cooling presented in CESSAR-DC Section 6.3.3.4 (Amendment N) credited only safety-grade systems (i.e., use the reactor coolant gas vent system instead of the auxiliary pressurizer spray for the pressure control) and assumed the SCS initiation to be consistent with the entry conditions for the System 80+ SCS design. The staff's evaluation discussed in this section is based on ABB-CE's reanalysis results. ABB-CE used NRC-approved methods (the LTC code) and credited only safety-grade systems in the reanalysis and demonstrated that the LTC operation strategy provides adequate core cooling without boric acid precipitation. The staff, therefore, concludes that the reanalysis is acceptable. On this basis, DSER Open Item 15.3.1-2 is resolved.

15.3.9 Steam Generator Tube Rupture (SGTR)

An SGTR event is a penetration of the barrier between the RCS and the main steam system. The event is caused by the failure of an SG tube. It is important to maintain the integrity of the barrier between the RCS and main steam system, from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected steam generator. Before turbine trip, the radioactivity is transported through the turbine to the condenser where noncondensable radioactive materials would be released by the condenser air ejector. Following the reactor trip and turbine trip, with the steam bypass system in its manual mode, the SG safety valves open to control the main steam system pressure. The operator can isolate the damaged SG after the reactor trips. The RCS and SG system can be cooled down by manual operation of the emergency feedwater and the atmospheric dump valves or the turbine bypass valves, and by using the unaffected SG. The analysis presented in CESSAR-DC Section 15.6.3 assumes that operator action is delayed until 30 minutes after the initiation of the event.

The radiological consequences for the SGTR transient, which are evaluated in Section 15.4 of this chapter, are also dependent on the break size. For break sizes resulting in a reactor trip, the initial leak rate decreases from the value equivalent to a double-end rupture, and the offsite dose also decreases because of the drop in the integral leak. The decrease in the break size also delays the time of a reactor trip. As the break size is decreased further, the integral leak is reduced and the radiological consequences will be less severe. Therefore, the worst break size is the largest assumed break of a full double-ended rupture of a steam generator tube.

Previously approved methods were used for the analysis. As discussed in Section 15.1 of this chapter, the computer codes used are: CESEC-III for calculating the system behavior, TORC for conducting the core thermal- hydraulic analysis, and the CE-1 correlation for determining the DNBR.

ABB-CE analyzed three SGTR events and presents the analysis in CESSAR-DC Sections 15.6.3.1 through 15.6.3.3, respectively:

- SGTR without a LOOP
- SGTR with a LOOP
- SGTR with a LOOP and a single failure

To be consistent with the assumption of no delay time of LOOP following a turbine trip discussed in Section 15.1 of this chapter, a LOOP was assumed to occur coincidently with a turbine trip for the cases of minimum DNBR calculations. However, for the cases of radiological release calculations, LOOP was assumed to occur 3 seconds after a turbine trip. A series of SGTR calculations showed that the calculated minimum DNBRs do not fall below 1.24 for cases with no or 3 seconds LOOP delay after a turbine trip. These analytical results indicated that no fuel failure will result from an SGTR event and the assumption of zero or 3 seconds LOOP delay time does not affect the radiological releases for an SGTR event. In the analysis, a limit of 1-gpm (3.8 L/min or 0.0038 m^3/min) leakage in the unaffected SG was assumed for duration of the transient. Various combinations of initial conditions were considered to maximize the primary releases to the atmosphere during the SGTR transient. The sensitivity study determined that the initial condition resulting in the worst radiological releases was a combination of the maximum RCS pressure, maximum pressurizer liquid volume, maximum SG liquid volume, maximum core power, maximum core coolant flow, and maximum core coolant inlet temperature. For the case of the maximum consequences of a radiological release, ABB-CE stated that the most limiting single failure is the failure of an atmospheric dump valve at the affected SG to close at 1,800 seconds after initiation of an SGTR event.

The SGTR analyses showed that the maximum RCS and secondary pressures do not exceed 110 percent of design pressure following an SGTR accident with and without a LOOP, thus assuring the integrity of the RCS and main steam system; the minimum DNBR is greater than the safety limit DNBR of 1.24, ensuring that no fuel failure will occur. The staff, therefore, concludes that the SGTR analysis is acceptable. The staff's evaluation of the radiological release appears in Section 15.4 of this chapter.

SGTR/Containment Bypass

In the DSER, the staff raised an issue concerning the potential for containment bypass due to a rupture of one or more SG tubes. During a tube rupture event, the potential exists for lifting of SG safety or relief valves and discharging primary system radioactive inventory outside the containment. Such a containment bypass is undesirable for either a design-basis event or a postulated severe accident. Consequently, the staff believes that possible mitigation of this containment challenge should be considered. This was designated as DSER Open Item 15.3.8-1.

In SECY-90-016, Issue III.D, "Containment Performance," the staff required ABB-CE to reduce the potential for conditional containment failure through use of quantitative guidelines or alternative deterministic objectives. In addition, with respect to design-basis events, in the URD, EPRI states that PWR containments should be designed to produce a leak-tight barrier to prevent uncontrolled release of radioactivity in the event of a postulated accident. Containment bypass due to SG tube ruptures would potentially violate containment integrity and hamper meeting both the severe-accident (SECY-90-016 "Evolutionary LWR Certification Issues and their Relationship to Current Regulatory Requirements," January 1990) and EPRI containment performance goals.

In the DSER and SECY-93-087, the staff stated that evolutionary PWR designers should consider potential design features that would reduce the amount of containment bypass leakage from such a scenario. Features that could mitigate the releases from a tube rupture may include:

- incorporating a highly reliable (closed-loop) SG, shell-side, heat removal system that relies on natural circulation and stored water sources

- piping some SG relief valve discharge back into the primary containment

- increasing the SG shell-side pressure capacity with a corresponding increase in the safety valve setpoints

ABB-CE should evaluate the potential benefit of such mitigation features. ABB-CE should consider mitigation features that would likely be available following a postulated severe accident. Rejection of any option should be justified on the basis of low risk, taking into account the uncertainties in these calculations. ABB-CE should incorporate appropriate revisions to the CESSAR-DC to reflect potential benefits. The staff continued to state in the DSER that it would expect the System 80+ design to include an assessment of such features and to address the desirability of this mitigation function.

In SECY-93-087 "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 1993, the staff recommended that the Commission approve the position to require that the evolutionary PWR designs assess features to mitigate the amount of containment bypass leakage that could result from SGTRs. In its July 21, 1993, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for SGTRs for the System 80+ design is as follows:

> The application for design certification must include an assessment of potential design improvements to mitigate the amount of containment bypass leakage that could result from SGTRs that are significant and practical and do not impact excessively on the plant. The application must also include a best-estimate, systematic evaluation of the plant response to an SGTR to identify potential design vulnerabilities.

To address DSER Open Item 15.3.8-1 and the proposed applicable regulation, CESSAR-DC Appendix 5F provides an evaluation of SGTR events for the System 80+ design. Consistent with the staff recommendations in SECY-90-016 and SECY-93-087, the staff, in Section 15.3.9 of this chapter, states that ABB-CE should consider potential design features that would reduce the amount of containment bypass leakage from an SGTR event. The staff also recommended three potential design features that could mitigate the release from the tube ruptures. Instead of these three design features, however, CESSAR-DC Appendix 5F provides ABB-CE's study to evaluate certain automatic design features that can be used to enable the plant to mitigate SGTR consequences. Sections 3 and 4 of CESSAR-DC Appendix 5F describe these analyses and provide an evaluation of the attendant benefits and limitations of each of these automatic design features. As noted in Section 19.2.3.3.5.2 of this report, ABB-CE assessed the three design alternatives identified in SECY-93-087 in a report dated September 23, 1993 and titled, "Design Alternatives for the System 80+ Nuclear Power Plant," and found these alternatives to be cost prohibitive (see Section 19.2.3.3.5.2 for more information).

As a result of these analyses, some features have been added to the System 80+ design to reduce the potential containment bypass leakage from the SGTR events. These features include: (1) a design modification to the component cooling water system (CCWS) to ensure continued cooling of the instrument air compressors after a safety injection actuation signal (SIAS), (2) addition of two nitrogen-16 (N-16) radiation monitors (one per SG) in the steamlines, (3) implementation of technical specifications and ITAACs related to N-16 monitors, and (4) emergency operations guidelines (EOGs) improvements. The staff has determined that this open item has been properly addressed with these enhancements. In arriving at this conclusion, the staff considered whether the System 80+ design provides sufficient time, diagnostic information, mitigation capability, and proper EOGs for operator coping actions following an SGTR event to mitigate the consequence. This evaluation is addressed below.

(1) <u>Lapse Time Before MSSV Challenge</u>

A primary consideration in the staff's review was the likelihood of a main steam safety valve (MSSV) lifting during an SGTR event and then failing to close. Such an occurrence results in an unisolable release bypassing the containment. Therefore, the evaluation in CESSAR-DC Appendix 5F includes determination of the time following a rupture of one to five SG tubes in the System 80+ design before an MSSV lifts, assuming no operator action. This will determine the time available for operators to take mitigative actions to keep the MSSV from lifting.

New design features have been incorporated into the System 80+ design relative to the System 80 design which extend the time available for operator action to prevent MSSV lifting following an SGTR event. The System 80+ turbine bypass system (TBS) with automatic actuation by the steam bypass control system (SBCS) directs steam from all bypass valves to the main condenser, unlike the System 80 design in which two of the turbine bypass valves release the steam directly to the atmosphere. In the event of an SGTR, the TBS will automatically dump steam to the condenser thereby relieving secondary pressure to reduce the possibility of reaching the MSSV setpoint. This feature, along with the fact that the System 80+ steam generators have larger secondary-side volume, extends the time before the MSSVs are challenged in an SGTR event without any operator mitigative action.

Realistic analyses were performed for a five-tube rupture case and a single-tube rupture case. These analyses took credit for certain non-safety-grade control systems and equipment. The analyses are described in Section 4 of Appendix 5F, and show that the reactor trips on hot-leg saturation at about 149 seconds and 1,289 seconds for the five-tube rupture and single-tube rupture cases, respectively. The steam bypass system is automatically actuated about 3 seconds after the reactor trips. Within about 16 seconds of the reactor trip, SIAS occurs on low RCS pressure, and eventually the safety injection flow increases to about balance the leak flow. The TBS will continue to release the steam to the main condenser and, therefore, the water level of the faulted steam generator will not reach the main steam isolation valve (MSIV) setpoint for about 26 minutes and more than 3 hours, respectively, for the five-tube and single-tube rupture cases. Subsequent to the steam generator isolation by MSIV closure, the MSSVs will actuate on high pressure in the steam generator. Therefore, unless the operator takes appropriate actions, the MSSVs will lift after about 4 hours for a single-tube rupture, and after 30 minutes for a five-tube double-ended guillotine rupture. The staff believes there is sufficient time for operators to diagnose the program and to take mitigative actions to prevent the MSSVs from lifting.

ABB-CE analysis assumed proper functioning of the SBCS and TBS throughout the SGTR events. However, in the original System 80+ design, the air compressors that supply the instrument air for operation of the turbine bypass valves (TBS/SBCS) were cooled by the CCWS from the nonessential CCW header, which was isolated upon an SIAS. Therefore, limited by the air receiver size, the instrument air pressure could only be maintained for about 15 minutes after an SIAS. To ensure that the TBS/SBCS will continue to function throughout the SGTR event, the System 80+ CCWS was modified so that the CCW will continue to cool the air compressors to ensure availability of the instrument air to maintain operation of the turbine bypass valves even after the SIAS. This is done by changing the CCWS so that the instrument air compressors will be cooled by the CCW flow from the essential safety class header, which is not isolated upon an SIAS, and will ensure continuous supply of CCW throughout the SGTR event.

(2) Scenario Diagnostics

In addition to the steamline area radiation monitors and sample and blowdown radiation monitors which were included in the original System 80+ design, two N-16 radiation monitors, one per steam generator, were added to assist in diagnosis of SGTR events. As N-16 has a very short half-life and is essentially nonexistent outside of nuclear reactors, detection of the high-energy N-16 gamma radiation in the secondary side of a PWR steam generator is a definite indicator of a primary to secondary leak. The N-16 monitors on the steamline afford a sensitive and specific indication of primary coolant leakage and give a more timely notification of an increase in leak rate; also they may detect precursors to an SGTR event, as well as specific indication of the affected steam generator. One of the design features of the System 80+ monitors is latching of the N-16 alarm signal to remind operators of the indication after the reactor trips.

ITAAC Section 2.8.2, "Main Steam Supply System," has been updated to ensure inclusion of the N-16 monitors in the plant. ITAAC Table 2.9.4-2 is also revised to add the radiation monitors in the main control room minimum inventory of alarms. In addition, ABB-CE has also added TS 3.3.10 for containment bypass instrumentation associated with steam generator tube rupture, that is, the main steamline radiation monitors, the steam generator blowdown monitors, and the new N-16 monitors. Limiting condition for operation 3.3.10 requires operability of the N-16 radiation monitors in each steam generator for power levels above 25 percent.

In Section 5.6.3 of CESSAR-DC Appendix 5F, ABB-CE also commits to a primary-to-secondary leakage monitoring program, designated as COL Action Item 5F-1. This program will address three specific scenarios: (a) low-level or slowly increasing primary-to-secondary leakage, (b) rapidly increasing primary-to-secondary leakage (as described in Information Notice (IN) 91-43, "Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate," July 1991 and IN-88-99, "Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage," December, 1988), and (c) steam generator tube rupture (without leak before break). This program will also address instrumentation setpoints and methodology for equipment (including N-16 monitors) used to detect steam generator tube leakage and ruptures commensurate with those scenarios.

In addition, the System 80+ design uses the Nuplex 80+ Advanced Control Complex. This system uses displays which take human abilities into account to provide continuous plant response and safety system information to help the operator evaluate events. The staff believes that this Nuplex 80+ control complex and the N-16 radiation monitors will give the operator reliable, timely, and specific diagnostics for an SGTR event.

(3) Other Mitigation Features

The System 80+ design also has many manually operated systems that can be used to mitigate SGTR events. There are two safety atmospheric dump valves (ADVs) on each steam generator that serve as a controllable alternative for relieving secondary pressure. Unlike, the MSSVs, these ADVs have isolation valves upstream to assure closure in the event of a stuck-open ADV. Each steam generator also has a large-capacity liquid blowdown system that can be used to release the secondary liquid to the condenser if the steam bypass system is inoperable.

In the primary RCS, there are several systems that can be used to depressurize the primary system. These include the main and auxiliary pressurizer spray systems, the CVCS charging and letdown system, a reactor coolant gas vent system (RCGVS), and throttling of safety injection pumps. In addition, the System 80+ design also has a rapid depressurization system (RDS), which discharges to the in-containment refueling water storage tank (IRWST). This RDS can be manually actuated by the operator to depressurize the primary system. In the case of a stuck-open MSSV, the RDS and the high-capacity steam generator blowdown system provide contingency options to be used to rapidly lower the primary and secondary pressures, and minimize the release through the stuck-open MSSV. The IRWST, which is both a source of safety injection water and a quench tank that confines blowdown fluid within the containment, contains more than one-half-million gallons of borated water and can be refilled through the CVCS from the boric acid storage tank. The large amount of borated water in the IRWST increases the long-term recovery probability for unisolable steam generator leakages by preventing depletion of borated safety injection water and core damage.

(4) Emergency Operations Guidelines (EOGs)

ABB-CE revised the System 80+ EOGs to be consistent with the addition of N-16 monitors for primary-secondary leakage detection and the availability of the RDS for primary system depressurization. The Functional Recovery Guidelines of the EOGs have been modified to add a new success path, PC-7, "RDS During SGTR." PC-7 describes the use of the RDS to maintain the RCS pressure less than 1,200 psia so that the affected steam generator MSSVs are more likely to reclose or remain closed. The SGTR Recovery Guidelines of the EOGs have been revised to (1) include N-16 radiation monitors to identify the faulted steam generator and (2) add a contingency action that directs the operator to enter the Functional Recovery Guideline and pressure control Success Path PC-7 if the RCS pressure cannot be maintained below 1,200 psia using main or auxiliary sprays, operation of charging and letdown, operation of RCGVS, and throttling of safety injection pumps. The staff concludes that the revised EOGs are consistent with system design and provide reasonable guidance for the mitigation of an SGTR event.

PRA Insights

The staff reviewed the System 80+ probabilistic risk assessment (PRA) performed by ABB-CE and concludes that (1) the unisolated SGTR sequences, that is, the SGTR initiating event with unisolable leak outside the containment (e.g., because of stuck-open MSSVs) is not a significant contributor to the core damage frequency and (2) although the unisolated SGTR events are a significant contributor to the offsite risk, the overall risk is very low compared to the current generation of operating plants. Important PRA insights summarized in Section 19.1.3.3.3 of this report indicate that most of the risk involves SGTR events in which RCS pressure control is not established, the faulted steam generator is not isolated, and the operator fails to replenish the IRWST inventory. Two COL action items are identified: (1) the operator action to isolate the faulted SG is identified as a "critical task" item, which will be considered as part of the detailed control room design process and the development of plant operating procedures and training program, and (2) the operator actions to align CVCS to refill the IRWST following an SGTR with containment breach is identified as a "non-critical task" item, which is important enough to be included in the functional task analysis to be performed by the COL applicant as part of the detailed control room design with the resulting indication and control requirements being incorporated in the availability verification activity.

It is noted that the isolation of the faulted steam generator has been incorporated in the SGTR Recovery Guidelines of the System 80+ EOGs. ABB-CE has also committed to revise the SGTR Recovery Guidelines for System 80+ to require the operators to monitor and maintain the IRWST by replenishment from available sources as necessary.

SGTR/Containment Bypass Summary

After reviewing the safety analysis, design features, ITAACs, TSs, emergency operations guidelines included in the System 80+ design certification application, and PRA insights, the staff concludes that there is a reasonable assurance that SGTR events pose no undue threat to the public health and safety and the System 80+ design satisfies the staff's proposed applicable regulation for SGTRs. On this basis, DSER Open Item 15.3.8-1 is resolved with (1) the commitment of COL Action Item 5F-1 on the primary-to-secondary leakage monitoring program described in Section 5.6.3, Appendix 5F, of CESSAR-DC, and (2) commitment of two other COL action items identified from the PRA insights related to isolation of the faulted steam generator and refill of the IRWST during an SGTR event.

SG Overfill

In RAI Q440.109, the staff asked ABB-CE to address the staff's concern about preventing SG overfilling during an SGTR event. In its response of May 8, 1992, ABB-CE stated that in the event of an SGTR, the affected SG water level is controlled by the following actions: (1) control the affected SG water by initiating or terminating the emergency feedwater, (2) minimize primary-to-secondary-side pressure difference by using the pressurizer gas vent valves and throttling the SI flow, and (3) control the SG water level by steaming the affected SG via the SG atmospheric dump valve in the event of loss of ac power. By letter dated December 18, 1992, and CESSAR-DC Section 15.6.3.3.3.1C (Amendment N), ABB-CE provided the technical basis for the SG overfill prevention, which is based on the results the SGTR analysis presented in CESSAR-DC Section 15.6.3.3 for the case with a loss of offsite power, and a stuck-open ADV. This analysis simulates the SGTR event for a period of 8 hours and incorporates the operator actions necessary to prevent the affected SG from overfilling. The analytical results (CESSAR-DC Figure 15.6.3-42B) show that, after about 4 hours, the break flow rate reduces from 15.9 kg/second (35 lb(mass)/sec) to 5.5 kg/second (12 lb(mass)/sec). At this time, the operator is expected to use SG level control to prevent SG overfilling. On the basis of critical flow of steam through one ADV at the steam pressure, a flow of about 32 kg/second (70 lb(mass)/sec) can be achieved. This calculated result shows that the break flow can be accommodated by partial opening of one ADV (17 to 50 percent) to maintain an essentially stable SG level. The effects on the radiological release by opening the ADV have been accounted for and results are presented in CESSAR-DC Table 15.6.3-9. Also, ABB-CE agreed in the response of November 27, 1991, to RAI 440.109 and CESSAR-DC Section 15.6.3.3.2-H that the System 80+ emergency operations guidelines will include a step (step 11.b of the steam generator tube rupture recovery guidelines) to prevent backfill from the secondary system through the affected SG by maintaining a positive pressure difference between the primary and secondary systems. This step is to prevent boron dilution during an SGTR. On the basis of this discussion and the staff's finding that the System 80+ EOGs contain appropriate steps for preventing SG overfilling, the staff concludes that the issue of SG overfilling prevention is adequately addressed. On this basis, DSER Open Item 15.3.8-2 is resolved.

15.3.10 Anticipated Transients Without Scram (ATWSs) (CESSAR-DC Section 19.4.13)

An ATWS event is an anticipated operational occurrence (such as loss of normal feedwater, loss of condenser vacuum, or LOOP) combined with an assumed failure of the reactor trip system (RTS) to shut down the reactor. On July 26, 1984, the staff amended the Code of Federal Regulations (CFR) to include 10 CFR 50.62, "Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (known as the "ATWS rule"). The ATWS rule, as amended on November 6, 1986, and April 3, 1989, requires nuclear power facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients, and to mitigate the consequences of an ATWS event.

In general, the equipment to be installed in accordance with the ATWS rule is required to be diverse from the existing RTS, and must be capable of being tested at power. This equipment is intended to provide needed diversity to reduce the potential for common-mode failures that could result in an ATWS leading to unacceptable plant conditions.

The basic requirements for the pressurized-water reactor manufactured by Combustion Engineering are specified in paragraphs (c)(1) and (c)(2) of 10 CFR 50.62 (ATWS rule) which state, in part:

> Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater and initiate a turbine trip under conditions indicative of an ATWS [paragraph (c)(1)] and must have a diverse scram system from the sensor output to interruption of power to the control rods [paragraph (c)(2)].

The System 80+ design includes a control-grade alternate protection system (APS) to provide an alternate reactor trip signal and an alternate emergency feedwater actuation signal separate and diverse from the safety-grade reactor trip system. The staff's review of ABB-CE's compliance with the ATWS rule appears in Sections 7.7.1.12 and 7.7.2 of this report.

The staff asked ABB-CE to submit an ATWS analysis demonstrating that the System 80+ ATWS response is within the bounds considered by the staff during the deliberations leading to the ATWS rule (10 CFR 50.62). In response, ABB-CE submitted the results of ATWS analyses (RAI Q440.111) and subsequently included them in CESSAR-DC Section 19.4.13 (Amendment S) for the staff review. These analyses were performed on a best-estimate basis. No credit was taken for reactor trip by the APS. ABB-CE's analytical results demonstrated that the maximum peak pressure (resulting from the limiting case, loss of main feedwater without a turbine trip) is less than

2.17×10^4 kPa (3,150 psia), and is within the bounds of ABB-CE's response considered by the staff for the ATWS rule making. The analysis was performed assuming a power level of 3,817 MWt. ABB-CE has subsequently increased the power level by 3 percent. ABB-CE did not, however, reanalyze the ATWS event at the increased power level. ABB-CE's basis for not providing a reanalysis is that the pressurizer safety valve capacity has also been increased by 14 percent. ABB-CE stated that the increase in safety valve capacity will offset the effect of the power increase. The staff agrees that the increase in safety valve capacity will act to limit the effect of increasing power. Since the System 80+ design complies with the requirements of the ATWS rule specified in 10 CFR 50.62, and the ATWS analysis shows that the maximum peak pressure is comparable to that considered for the ATWS rule making, the staff concludes that the System 80+ design satisfactorily addresses the ATWS concerns and is acceptable. The staff's evaluation of the alternate protection system (APS) appears in Section 7.7.1.12 of this report.

The original TSs for the System 80+ design allow positive MTCs for operation of System 80+ plants. A positive MTC design is not consistent with the requirements of EPRI's URD, which requires a negative MTC design. Further-more, ATWS analysis submitted to the staff assumed a negative MTC to calculate the peak pressure to be within the acceptable bound considered for the ATWS rule making. To address the staff's concern, ABB-CE revised TS 3.1.4 (Amendment K) to limit MTCs to be nonpositive. Since this approach is consistent with the assumption used in the ATWS analysis, the staff determines that it is acceptable. (Also see the staff evaluation of the positive MTC discussed in Section 4.3.2 of this report).

15.3.11 Liquid Tank Failure Accident

See Section 15.4.2.6.2 of this chapter for a discussion of a postulated radioactive release due to liquid containing-tank failures.

15.3.12 Conclusions

ABB-CE has presented results for various transient and accidents that conform to the acceptance criteria as detailed in SRP Chapter 15. ABB-CE has submitted acceptable analyses to demonstrate adequate protection systems to mitigate design-basis transients and accidents in compliance with the applicable GDC relating to core coolability, control rod insertability, and primary and secondary system pressure boundary integrity.

15.4 Radiological Consequences of Design-Basis Accidents

15.4.1 General

The staff has reviewed the analyses prepared by ABB-CE related to the radiological consequences of abnormal operating transients as well as a broad spectrum of postulated accidents.

This section evaluates ABB-CE's analysis of the radiological consequences of such accidents. The following accident categories were considered:

- increase in heat removal by the secondary system
- decrease in heat removal by the secondary system
- decrease in the reactor coolant flow rate
- reactivity and power distribution anomalies
- decrease in RCS inventory
- release of radioactive material from a subsystem or component

ABB-CE indicated its intent to incorporate appropriate aspects of the revised source term for the System 80+ design. By letter dated October 19, 1992, the staff described the specific information to be submitted by ABB-CE to enable the staff to complete its review.

In this letter, the staff noted that, with the issuance of (draft) NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," June 1992, for public comment, the staff believes that future advanced light water reactors (ALWRs) should utilize the revised source term since it reflects the NRC's most current understanding of fission product behavior following severe reactor accidents.

By letter dated March 26, 1993, ABB-CE submitted radiological dose predictions generated using the new, physically-based source term and assuming a normal containment leakage rate of 0.5 percent volume/day.

Within the broad spectrum of accidents postulated above, a number of postulated events/accidents were considered. These postulated event/accidents are representative of the range of events involving the various engineered safety feature systems and components and ABB-CE analyzed them in detail. The staff has independently analyzed the radiological consequences of these postulated accidents as described in the discussion that follows.

In implementing the new physically based radiological source-term model, ABB-CE applied this model to design-basis safety analysis dose calculations and assessed the resulting design impacts on the necessity for safety-grade charcoal filters as well as the impact on equipment qualification.

ABB-CE considered four topics in evaluating the impact of a revised, physically based source term on the System 80+ design. First, ABB-CE considered a revised fission product release to the containment; second, ABB-CE considered a revised fission product transport and deposition model; third, ABB-CE considered a revised dose consequence model; and finally, ABB-CE considered the impacts of the revised source term on the issue of the qualification of safety-related electrical equipment.

In evaluating the impact of the new source term on dose consequences, ABB-CE considered such items as source term modeling (radionuclide release to the environment), intake-to-dose conversion factors, bounding χ/Q values, computer code structure, and the results of the dose analysis. For additional information on source term related technical and licensing issues, refer to the appendix to this chapter, Appendix 15A.

As documented in Section 15.1 of this report, the staff found that application of the convolution method to the System 80+ design is within the applicable limits of the approved calculational method and was acceptable for the System 80+ fuel type. As noted in Section 15.1, DSER Open Item 15.1-2 is resolved.

15.4.2 Accident/Event Categories Considered by ABB-CE

15.4.2.1 Increase in Heat Removal by the Secondary System: Main Steamline Break (MSLB)

ABB-CE considered a variety of cases for a main steamline break to find the maximum potential for dose at the exclusion area boundary. The cases considered by ABB-CE to maximize the degradation in fuel performance and dose at the site exclusion area boundary include:

- steamline break outside the containment, upstream of the main steam isolation valve during full-power operation with a loss of offsite power (LOOP), reactor/turbine trip, and maximum Technical Specifications (TS) SG tube leakage
- steamline break outside the containment, upstream of the MSIV during zero power operation with a concurrent LOOP in combination with maximum TS SG tube leakage of 3.79 L/min (1.0 gpm).

In analyzing the consequences of main steamline failures, ABB-CE determined that the offsite thyroid doses for the steamline break outside the containment during full-power operation concurrent with a LOOP and reactor/turbine trip was the limiting main steamline-break case.

The staff reviewed ABB-CE's analysis of the radiological consequences of an MSLB and verified that it was performed using appropriate regulatory positions. For the case of the limiting steamline break which ABB-CE conservatively assumed was a 0.5-percent fuel failure, the staff reviewed ABB-CE's methodology and assumptions and found that this analysis was performed using appropriate regulatory guidance and positions as outlined in the SRP for the case of a steamline break outside the containment with a LOOP, reactor/turbine trip, stuck CEA, and maximum TS-allowable primary to secondary leakage.

On the basis of its review, the staff concludes that the calculated radiological consequences of a postulated main steamline failure outside the containment do not exceed (1) the exposure guideline values given in 10 CFR Part 100 for both the preaccident iodine spike and fuel failure cases and (2) a small fraction (10 percent) of these exposure guidelines for the event-generated iodine spike case. Consequently, the staff finds the System 80+ design acceptable with respect to the radiological consequences of a main steamline failure outside the containment.

As discussed in Section 15.1 of this chapter, ABB-CE has agreed not to take credit for a three second LOOP delay in the transient and accident analysis. On this basis, DSER Open Item 15.4.1.1-1 is resolved.

In addition, the staff noted in DSER Open Item 15.4.1.1-2 that ABB-CE did not analyze the radiological consequences of increases in heat removal by the secondary system. However, the SRP does not require such analysis for non-fuel failure events since the radiological consequences would be minimal. On this basis, DSER Open Item 15.4.1.1-2 is resolved.

As discussed in Section 15.1 of this chapter, the staff found that application of the convolution method to the System 80+ design is within the allowable limits of the approved calculational method and was acceptable for the System 80+ fuel type. On this basis, DSER Open Item 15.4.2.1-1 is resolved.

15.4.2.2 Decrease in Heat Removal by the Secondary System: Feedwater System Pipe Breaks

Of the many events that could lead to a decrease in heat removal by the secondary system, only one, a feedwater system pipe break, was judged to have potential offsite radiological consequences associated with it. The limiting feedwater line break (FLB) event occurs with a break downstream of the check valves, inoperability of the main feedwater system (MFS), and low enthalpy break discharges. The resultant loss of feedwater flow to both steam generators results in a reduction in steam generator water levels and increasing steam generator temperatures.

In conducting the evaluation of this event to identify the limiting break size, ABB-CE considered a spectrum of postulated break sizes and concluded the limiting break size for maximum peak RCS pressure is 0.056 m^2 (0.6 ft^2). ABB-CE determined that the minimum DNBR experienced throughout the event is less than 1.24, and that less than 0.15 percent fuel failure would result. DNBR is minimized at a break size of 0.02 m^2 (0.2 ft^2) which results 0.22 percent fuel failure. A total of 65,900 kg (145,000 lb(mass)) of steam was calculated to be released from the feedwater system to the atmosphere during the first 30 minutes of the transient with a decontamination factor of 1 for the release from the affected steam generator and DF of 100 for the unaffected SG. In addition, a total of 79,450 kg (175,000 lbm) of affected steam generator mass inventory is released into the containment via the break with a DF of 1. During the period between 30 minutes and 8 hours, ABB-CE assumed that steam releases are the same as for the steamline break case, since the cooldown is the same.

ABB-CE considered two sources of activity in analyzing the radiological impact of this event: the initial steam generator inventory activity and activity added to the secondary side from primary to secondary tube leaks. TS activity limits in both the primary and secondary side were assumed. In addition to the TS reactor cooler activity, the primary side activity includes the gap activities of the failed fuel. In addition to the minimum DNBR case, ABB-CE also analyzes an overpressure case in which a pre-accident iodine spike or an event-generates iodine spike is assumed. In ABB-CE's analysis, the worst case thyroid dose at the exclusion area boundary is 0.2 Sv (20 rem). ABB-CE also computed a whole-body dose of 3.3 x 10^{-4} Sv (0.033 rem) at the exclusion area boundary.

ABB-CE noted that both the RCS and main steam pressure boundaries remain intact, and that maximum calculated doses do not exceed a small fraction of 10 CFR Part 100 guideline values. The staff has reviewed ABB-CE's calculation of the offsite dose consequences (to the whole body and the thyroid) based upon the mass releases reported by ABB-CE and a conservative description of the plant response to the accident. A χ/Q value of 1.0 x 10^{-3} sec/m^3 for the 0-2-hour time period was used in the evaluation of the radiological consequences of a feedwater-line break event. The staff concluded that the TS limits on primary and secondary coolant activities will limit potential offsite doses to values which are less than a small fraction of the exposure guideline values of 10 CFR Part 100. Therefore, the calculated offsite dose consequences of a feedwater-line break are within the acceptance criteria given in SRP Section 15.2.8 and are acceptable.

As discussed in Section 15.1 of this report, the staff found that application of the convolution method to the System 80+ design is within the applicable limits of the approved calculational method and was acceptable for the System 80+ fuel type. In addition, since no fuel failure is expected from a loss of condenser vacuum event, no radiological consequence analysis is required. On this basis, DSER Open Item 15.4.1.2-1 is resolved.

15.4.2.3 Reactor Coolant Flow Rate Decrease: Single Reactor Coolant Pump Rotor Seizure With Loss of Offsite Power

For such events as the seizure of a reactor coolant pump rotor, the major area of concern is the minimum hot channel DNBR. The DNBR determines whether a fuel design limit has been exceeded and, therefore, whether fuel damage can be expected to occur. As documented is Section 15.1 of this chapter, the staff found that application of the convolution method to the System 80+ design is within the applicable limits of the approved calculational method and was acceptable for the System 80+ fuel type. On this basis, DSER Open Item 15.4.2.3-1 is resolved.

In performing its analysis related to this event, ABB-CE concluded, after considering a number of single failures which could affect RCS behavior during the first 4 seconds of this transient, that none of the failures considered would lead to a more adverse transient DNBR limit than that predicted for the single reactor coolant pump rotor seizure event. However, calculated radiological consequences are maximized by considering steam releases through the main steam safety valves (MSSVs) and a single active failure of an atmospheric dump valve (ADV) to close in combination with a pump rotor seizure event with a LOOP.

In analyzing the radiological consequences of the single reactor coolant pump rotor seizure with a LOOP, ABB-CE assumed that condenser cooling water was not available for the duration of the transient and that for the first 30 minutes of the transient, cooldown is accomplished utilizing the main steam safety values. Operator action to actuate the ADVs is assumed at 30 minutes, and one ADV is assumed to stick open during the next 30 minutes, when the operator is assumed to close the ADV block valve.

In ABB-CE's analysis of the radiological consequences of this transient, 1.2 percent of the fuel was calculated to experience DNB and was, therefore, assumed to have failed. The staff has completed its review of ABB-CE's evaluation of the radiological consequences of a locked rotor event assuming 1.2 percent failed fuel and for the event-generated iodine spike case. The results of this evaluation indicate that radiological consequences of a locked rotor event are within staff acceptance criteria as given in SRP Section 15.3.3 (i.e., that activity releases are such that calculated doses at the exclusion area boundary are less than a small fraction of 10 CFR Part 100 guideline values). In conducting its evaluation on the radiological consequences of a locked rotor event, ABB-CE utilized appropriate guidance from SRP Section 15.3.3 as modified by applicable assumptions made in draft NUREG-1465. In this regard, ABB-CE assumed gap fractions for relevant isotopes (noble gases, iodines, cesiums, and rubidiums) consistent with draft NUREG-1465. Additionally, ABB-CE assumed chemical species of iodines in the gap based on draft NUREG-1465 (namely, 95-percent particulate, 4.75-percent elemental, and 0.25-percent organic). ABB-CE also conservatively assumed that the gap activity of the failed fuel is released instantaneously at the time of the accident. Therefore, the System 80+ design is acceptable with respect to the locked rotor transient. For additional information on source term related technical and licensing issues, refer to the appendix to this chapter.

15.4.2.4 Reactivity and Power Distribution Anomalies: Control Element Assembly Ejection Accidents

ABB-CE postulated and analyzed the effects of a control element assembly (CEA) ejection accident in which a circumferential rupture of the control element drive mechanism (CEDM) housing of the CEDM nozzle occurred. For this evaluation, ABB-CE considered a spectrum of initial power conditions to determine the limiting case for this transient.

As documented in Section 15.1 of this report, the staff found that application of the convolution method to the System 80+ design is within the allowable limits of the approved calculational method and was acceptable for the System 80+ fuel type. On this basis, DSER Open Item 15.4.1.4-1 is resolved.

ABB-CE determined that the greatest potential for offsite dose consequences for this event was the case initiated from hot, full-power conditions. This case was determined to have the greatest potential for postulated fuel failures and offsite dose consequences.

The ruptured CEDM pressure housing is assumed to release activity immediately to the containment and instantaneous mixing throughout the containment is assumed. In the analysis of the radiological consequences of a CEA ejection accident, ABB-CE noted that ejection of a CEA causes core power to increase rapidly due to the prompt positive reactivity insertion or addition. ABB-CE noted in its analysis that following a postulated CEA ejection event, 6.8 percent of the fuel is calculated to experience DNB. ABB-CE assumed in its analysis that two sources of offsite radiation exposures would occur, namely, the activity available for leakage from the containment and the activity released from the main steam safety valves and the atmospheric dump valves during cooldown. In performing its analysis, ABB-CE utilized the assumptions from RG 1.77, Appendix B as modified by NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," June 1992. Specifically, ABB-CE considered the activity in the fuel pellet cladding gap to be composed of 5 percent of the core iodine, 5 percent of the core noble gas, and 5 percent of the core cesium/rubidium fuel inventory at the end of core life. This inventory was developed by assuming continuous maximum full-power operation. In addition, ABB-CE assumed that for those fuel pins that are predicted to experience DNB, all of the activity in the pellet cladding gap is assumed to be instantaneously mixed throughout the containment and available for leakage to the atmosphere.

In addition, ABB-CE also considered activity released from the secondary system following the CEA ejection event. This activity was assumed to consist of activity initially in the steam generators plus additional secondary-side activity arising from primary to secondary leakage of the TS reactor coolant activity and the released failed fuel gap activity at the maximum rate allowed by TS. The total dose to the maximum-exposed individual is given by the greater of the containment leakage component and the primary to secondary leakage component. ABB-CE determined a thyroid dose for this event of 0.70 Sv (70 rem) via the containment pathway or 0.17 Sv (17 rem) via the secondary pathway.

The staff reviewed ABB-CE's analysis of the radiological consequences of a control element assembly ejection accident using the assumptions specified in NUREG-1465 and finds that the analyzed radiological consequences of this event are within the acceptance criteria of SRP Section 15.4.8. The staff concludes that the site parameters specified with respect to acceptable site atmospheric dispersion characteristics and minimum exclusion area and low population zone distances, in conjunction with the System 80+ design, are sufficient to give reasonable assurance that the calculated radiological consequences are well within the exposure guidelines as set forth in 10 CFR 100.11. In ABB-CE's analysis, it was assumed that containment sprays were not operating and that the activity in the fuel pellet/cladding gap is composed of 5 percent of the core iodines, 5 percent of the noble gases, and 5 percent of the cesium and rubidium in the fuel at the end of core life. In addition, ABB-CE's analysis took credit for the filtration capability of the annulus ventilation capability only after 30 minutes. ABB-CE also conservatively assumed that the gap activity of the failed fuel is released instantaneously at the time of the accident. For additional information on source term related technical and licensing issues, refer the appendix to this chapter.

15.4.2.5 Decrease in Reactor Coolant System Inventory

15.4.2.5.1 Double Ended Break of a Letdown Line Outside Containment

ABB-CE selected for analysis the double-ended break of the letdown line outside of the containment (upstream of the letdown line control valve), because it is the largest line. Consequently, failure of this line results in the largest release of reactor coolant outside the containment.

In performing its analysis, ABB-CE did not consider a single active failure of an isolation valve to close, because the letdown line includes three isolation valves situated in series inside the containment.

ABB-CE stated that 12.3 kg/second (27 lb(mass)/sec) of primary coolant is released as a result of a double ended break of a letdown line outside the containment, upstream of the letdown line control valve. In addition, ABB-CE noted that the maximum break flow, which is about 1.5 times the expected letdown flow, is limited to 12.3 kg/second (27 lb(mass)/sec) by the use of letdown line orifices located inside the containment downstream of the letdown line heat exchanger. ABB-CE assumed a decontamination factor (DF) of 1 for the nuclear annex (i.e., no credit was taken for retention or filtration of iodine in the escaped fluid). In the CESSAR-DC, ABB-CE noted that the letdown line orifices are provided inside containment, which limit the leakage of reactor coolant outside containment to a value such that regulatory acceptance criteria from SRP 15.6.1 are satisfied. On this basis, DSER Open Item 15.4.2.4-1 is resolved.

ABB-CE also assumed that 19.8 percent of the escaping fluid flashed to steam, based on the fraction of primary fluid that flashes to steam in the nuclear annex. This fraction of escaping fluid that flashes to steam in the nuclear annex is based on the enthalpy of the escaping fluid. ABB-CE also took no credit for ground deposition or radioactive decay of activity that escapes to the exclusion area boundary.

Further, ABB-CE assumed that the pressurizer level control system failed such that the charging flow rate was maximized, thereby causing higher break flow rates during the transient and maximizing the radiological consequences of this transient.

The staff has reviewed ABB-CE's analyses of the radiological consequences of the failure of a letdown line outside the containment and concludes that with the specified site parameter acceptance criteria, the System 80+ design is sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated small-line failure outside the containment in combination with an event-generated iodine spike, do not exceed a small fraction of the exposure guideline values stated in 10 CFR Part 100.

15.4.2.5.2 Steam Generator Tube Rupture

Steam generator tube rupture (SGTR) events involve a sudden failure of a steam generator U-tube, which provides a barrier between the RCS and the main steam system. In the normal course of this event, radioactive material from the leaking steam generator tube mixes with the shell-side water in the affected steam generator. In analyzing the radiological consequences of an SGTR, ABB-CE considered the following three different event sequences:

(1) SGTR without a concurrent loss of offsite power
(2) SGTR with a concurrent loss of offsite power
(3) SGTR with a loss of offsite power and a single failure

Because no fuel failure is expected to occur as a result of an SGTR event under any of these conditions, ABB-CE assumed a 3-second time delay between the turbine trip and the loss of offsite power.

ABB-CE also calculated that the minimum DNBR stayed above the specified acceptable fuel design limit of 1.24 throughout the SGTR event for each of the cases considered. Consequently, as noted above, no fuel failure is predicted to occur for any of the SGTR events analyzed.

An SGTR results in a reactor and turbine trip, a main steam pressure increase, and opening of the main steam safety valves to control main steam system pressure. Venting continues via this pathway from the affected steam generator until the secondary-side pressure is below the main steam safety valve setpoint.

ABB-CE further assumed after 1,800 seconds (30 minutes) the operator initiates a plant cooldown using the unaffected steam generator, atmospheric dump valves, and the emergency feedwater system. In the ABB-CE analysis, it was assumed that for releases from the unaffected steam generator, a DF of 100 resulted for the iodines.

The staff reviewed ABB-CE's analysis of the radiological consequences of an SGTR event with a LOOP and a limiting single failure. The limiting single failure was determined to be a stuck-open ADV. Failure of the ADV to close in the affected steam generator after the operator initially opens it results in additional steam release until the operator is able to isolate the ADV by closing the associated block valve. The staff concludes that the site parameters selected, with respect to the exclusion area boundary and the low population zone, are sufficient to provide reasonable assurance that the calculated radiological consequences of an SGTR accident do not exceed the exposure guideline values given in 10 CFR Part 100 and (with the exception of event sequence 3) 10 percent of these exposure guideline values for an SGTR with an equilibrium iodine concentration in combination with an assumed accident-generated iodine spike.

15.4.2.5.3 Spectrum of Loss-of-Coolant Accidents (LOCAs) Resulting From Postulated Piping Failures

In analyzing the radiological consequences of the spectrum of LOCAs in the CESSAR-DC, ABB-CE utilized the assumptions made in RG 1.4 and in SRP Section 15.6.5, Appendices A and B (NUREG-0800) as modified by draft NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants," June 1992. The models used by ABB-CE in performing these analyses appear in Appendix 15A to the CESSAR-DC.

Draft NUREG-1465 gives the release magnitudes for the gap release and early in-vessel release phases of the accident. These release magnitudes are reproduced in Table 15.A-1 of the appendix to this chapter as fractions of the total core inventory.

ABB-CE assumed that releases were uniform over the duration given in Table 3.6 of draft NUREG-1465.

Consistent with draft NUREG-1465, ABB-CE assumed the entire release was in particulate form except for the noble gases and 5 percent of the iodines. For the purposes of radiological analyses, ABB-CE conservatively assumed that 0.25 percent of the iodine released was organic.

Containment sprays were assumed to operate to remove airborne radionuclides; these removed radionuclides are assumed to mix with the in-containment refueling water storage tank (IRWST) inventory.

Circulation of this liquid through various safety pumps and leakage through pumps seals and valves results in activity in various ESF rooms which vents to the atmosphere. Finally, ABB-CE took no credit for either radioactive decay in transit or for ground deposition in transit.

Control room operators receive radiation doses as a result of control room air intake and in-leakage of radioactive material; additionally, radiation doses are received at various offsite locations due to radionuclide dispersal from several sources. These sources include:

- Discharge of iodine spike activity contained in the reactor coolant.
- Direct containment leakage as well as filtered discharge from the containment annulus ventilation system. In calculating the radiological impact of the direct containment leakage, ABB-CE assumed containment leakage at the maximum value allowed by TS. ABB-CE considered the effect of the containment annulus ventilation system in filtering discharge via this pathway to the outside atmosphere. ABB-CE assumed a 10-percent bypass of the annulus ventilation system in performing its analysis.
- Discharge from the emergency safeguards features rooms. Radioactive materials migrate from the IRWST into the ESF rooms through leaks in pump seals and valves. These materials enter the ESF room atmosphere and are then discharged through filters to the outside atmosphere.

ABB-CE considered that releases of radioactive materials from the primary system were divided into three release phases: (1) coolant, (2) gap, and (3) early in-vessel.

In calculating the radiological impact of this accident, ABB-CE assumed the releases were uniform over the duration of the release.

For purposes of this evaluation, ABB-CE noted that the 2-hour exclusion area boundary dose and the 30-day low population zone dose are calculated from the start of the gap release.

On the basis of information in draft NUREG-1465, ABB-CE assumed that the entire release was particulate (except for the noble gases and 5 percent of the iodines). Of this 5 percent, 5 percent was assumed to be organic. ABB-CE also considered timing for releases arising from this accident consistent with draft NUREG-1465 and as given in Section 15.A.7 of the appendix to this chapter. For additional information on source term related technical and licensing issues, refer to the appendix to this chapter.

ABB-CE computed doses at the exclusion area boundary for releases during the first 2 hours and at the low population zone from releases over the assumed 30-day duration of the event. ABB-CE computed total doses at a given location by considering releases from the following release paths: (1) discharge through the containment power purge line before it is closed, (2) containment leakage and annulus ventilation system discharge, and (3) ESF rooms discharge.

In its analysis, ABB-CE selected and analyzed a design-basis LOCA and determined that the total radiological consequences of such an accident conform to the exposure guidelines of 10 CFR 100.11 with respect to the adequacy of the minimum distances specified to the exclusion area boundary and the low population zone. ABB-CE analyzed appropriate radionuclide sources and transport paths as described above.

The staff also reviewed ABB-CE's analysis of the radiological consequences of a LOCA to an individual at the low population zone boundary and concludes that the analysis was performed using staff-approved methodologies and assumptions. ABB-CE's analysis of the radiological consequences of a design-basis LOCA shows that the criteria of 10 CFR 100.11 are satisfied with respect to both the exclusion area boundary and the low population zone.

On the basis of its review of the methods, assumptions, and parameter definitions used by ABB-CE, the staff concludes that the System 80+ design is acceptable with respect to the radiological consequences of the design-basis LOCA.

15.4.2.6 Release of Radioactive Materials From a Subsystem or Component

15.4.2.6.1 Fuel Handling Accident

In analyzing the radiological consequences of a fuel handling accident (FHA), ABB-CE considered the dropping of a single fuel assembly during fuel handling.
ABB-CE also noted that heavy loads are restricted to preassigned travel zones and that they are not carried over stored fuel assemblies. ABB-CE further noted that equipment interlocks and procedures are also used to ensure that heavy loads are moved as planned.

ABB-CE analyzed the radiological consequences of an FHA occurring in the containment and an FHA occurring in the fuel building. In performing its analysis, ABB-CE assumed that the containment purge ventilation system was operating and that associated filters were in place during a postulated FHA inside the containment. Likewise, a similar accident inside the fuel building assumed release through the fuel building ventilation system and its assorted filters.

ABB-CE performed analyses to determine the maximum expected number of fuel rods calculated to fail as a result of a dropped fuel assembly; however, for purposes of analyzing the radiological consequences of this accident, ABB-CE assumed the failure of all 236 fuel rods in one spent fuel assembly at 72 hours after shutdown.

ABB-CE calculated offsite radiological consequences to the whole body from immersion and to the thyroid due to inhalation for the 0-2-hour time period at the exclusion area boundary (EAB) and for the 0-8 hour time period at the low population zone (LPZ) outer boundary. The staff finds that ABB-CE has provided an adequate system to mitigate the radiological consequences of a postulated fuel handling accident inside the containment and in the fuel building.

The staff concludes that the specified site parameters related to the EAB and LPZ, in conjunction with the operation of dose-mitigating engineered safety features and appropriate plant procedures, are sufficient to provide a reasonable assurance that the calculated offsite radiological consequences from a postulated FHA are well within the exposure guidelines of 10 CFR Part 100.

This conclusion is based on (1) the staff's determination that the plant design features and proposed procedural controls comply with the requirements of GDC 61 with respect to radioactivity controls; (2) the staff review of ABB-CE's assumptions and analyses of the radiological consequences from the fuel handling accident; and (3) the staff's review of ABB-CE's analyses using the assumption in RG 1.25, Positions C.1.a through C.1.k, and in the appendix to this chapter. In the analysis of the radiological consequences of an FHA, ABB-CE utilized the gap release fractions specified in Table 3.12 of draft NUREG-1465 and listed in Table 15.A-1 of the appendix to this chapter. For additional information on source term related technical and licensing issues, refer to the appendix to this chapter.

15.4.2.6.2 Postulated Radioactive Release Due to Liquid-Containing Tank Failures

In considering the postulated failures of tanks located outside the containment which could contain radioactive materials, ABB-CE considered as the most limiting tank failure the uncontrolled release of liquid from the boric acid storage tank (BAST). This tank, part of the chemical and volume control system (CVCS), is an ASME Code Section III, Safety Class 3, seismic Category I tank. Although the contents of the tank would be contained by a seismically-designed dike should a tank fail because of a seismic event, ABB-CE takes no credit for the dike per SRP Section 15.7.3, because it is not lined with stainless steel.

ABB-CE characterized this event as a rapid release of the BAST contents to the environment caused by seismically-induced failure of the tank. The liquid was considered to be released through cracks in the basin surrounding the tank. The liquid was released to the plant discharge where it was diluted before it reached the potable water supply. The concentration at the nearest potable water supply was equated to the limiting effluent concentration for each radionuclide.

The concentration of radionuclides in the BAST was specified as a fraction of the primary coolant concentration (PCC). For conservatism, ABB-CE took no additional credit for radioactive decay during either the purification process or transport to the nearest potable water supply. Per SRP Section 15.7.3, 80 percent of the volume is assumed to be released. Additionally, all radionuclides were assumed to be in the insoluble form.

Using an iterative process, ABB-CE concluded that the maximum allowable dilution factor is 2.55 x 10^{-6}. This value reflects the minimum extent to which the radioactive liquid released from the failed BAST will be diluted before reaching the potable water supply. On the basis of its review, the staff finds that the methodology and approach used by ABB-CE to establish a site acceptance criterion for the minimum dilution flow required to limit the concentration of radioactive material at the nearest potable water supply to values less than the effluent concentrations specified in 10 CFR Part 20, are acceptable.

15.4.2.6.3 Spent Fuel Cask Drop Accidents

SRP Section 15.7.5, "Spent Fuel Cask Drop Accident," specifies that if the potential drop during handling of a loaded cask is less than 30 feet, and if the handling procedures comply with all applicable criteria, then the radiological consequences of a spent fuel cask drop accident need not be estimated.

In the CESSAR-DC, ABB-CE notes that all cask lifts from the cask laydown area are limited to less than 30 feet. In addition, ABB-CE notes that the spent fuel cask handling crane operating procedures for handling spent fuel casks establish requirements for operator training, crane inspection, and approved cask handling procedures.

Finally, ABB-CE notes that the crane for handling casks has mechanical stops and electrical interlocks to prevent its movement over the spent fuel pool after the pool contains irradiated fuel.

Therefore, since plant design criteria and cask handling procedures conform to the applicable criteria of SRP Section 15.7.5, an evaluation of the radiological impact of a cask handling accident is not required.

15.4.3 Environmental Protection Agency (EPA) Protective Action Guideline (PAG) Dose Calculations

In Chapter 15 of the CESSAR-DC, ABB-CE presents the results of a dose calculation for a sequence which conservatively represents the systems and equipment availability for the majority of the core damage sequences in the System 80+ design. The source term for this sequence is, therefore, expected to be in agreement with most core damage sequences.

In Section 15.6.5.4 of the CESSAR-DC, ABB-CE presents the assumptions, methodology, and results of its dose calculations for a "PAG evaluation" of a LOCA, and compares them with the EPA PAGs. The assumptions include a severe accident with a large release of radioactivity from the core to the containment, an intact containment, and the containment sprays operational. The release is postulated to occur from the containment into the annulus at the design-basis leakage rate, through the annulus filters into the environment. Ten percent of the containment leakage is postulated to bypass the annulus. The calculations use the "best-estimate" approach. ABB-CE concludes that under the postulated event, the EPA PAG limits of 1 rem committed effective dose equivalent (CEDE) and 5 rem to the thyroid are met.

15.4.4 Evaluation

The NRC's radiological protection review of the System 80+ design is based on DBA doses, as discussed in Section 15.4.2 of this report. This evaluation of PAG doses has no bearing on the NRC's design basis safety conclusions and does not represent a change in the approach to emergency planning.

The staff used the applicable emergency planning (EP) regulations and the guidance in NUREG-0396, NUREG-0654, and "EPA Manual of Protective Action Guides and Protective Actions For Nuclear Incidents" (EPA Manual) for this evaluation. In NUREG-0396, the staff stated that "PAGs represent only trigger levels and are not intended to represent acceptable dose levels. PAGs are tools to be used as a decision aid in the actual response situation." As such, the staff notes that it is unnecessary to treat the plume PAGs as limits.

The planning basis for EP, as stated in NUREG-0396, notes that "a spectrum of accidents should be considered in developing a basis for emergency planning." Furthermore, NUREG-0396 states that "both the design-basis accidents and less severe core melt accidents should be considered when selecting a basis for planning pre-determined protective actions and that certain features of the more severe core melt accidents should be considered in planning to assure that some capability exists to reduce the consequences of even the most severe accidents." As noted above, the sequence used in this calculation represents the systems and equipment availability for the majority of the core damage sequence. While the source term used is expected to be in approximate agreement with most core damage sequences, the CESSAR-DC PAG evaluation is based on a single accident sequence.

The assumptions used in calculating the doses that could result from the DBA LOCA are conservative. The assumptions used in the dose calculations for comparison to PAGs, however, are best-estimates; hence the doses that would result would be much lower than those from postulated DBA LOCA calculations. However, as presented in NUREG-0396, the NRC's position has been that a spectrum of postulated conditions be considered in emergency planning, including harsh meteorological conditions.

In the CESSAR-DC, ABB-CE considers the dose from inhalation of radioactive material, from immersion, and from ground contamination, and expresses the dose using the CEDE concept. The exposure pathways considered in the CESSAR-DC are consistent with those recommended in the EPA Manual for the plume phase, except for the exposure time to ground contamination. In the CESSAR-DC, ABB-CE does not consider the doses from the ingestion pathway or from long-term ground exposure. In NUREG-0396, that staff states that "much lower releases of radioiodine could result in projected doses in excess of the ingestion PAGs without there being a potential to exceed plume exposure PAGs."

15.4.5 Conclusion

Although the staff did not perform independent calculations, it concludes that for the single-accident sequence postulated by ABB-CE, which bounds most severe accidents presented in CESSAR-DC, and using the best-estimate approach, the models used and assumptions made by ABB-CE provide results that are generally reasonable. However, as indicated above, ABB-CE's approach does not consider the effects of a spectrum of accidents which are the basis for the NRC's EP regulations.

Appendix 15A

The Application of Source-Term Issues Described in a Draft Commission Paper[1] to the System 80+ Design

15A.1 General

This appendix addresses source term related technical and licensing issues pertaining to the System 80+ design. Significant technical positions relative to the implementation of the new accident source term for such evolutionary designs as the System 80+ design are addressed as applicable. The staff has determined that in its evaluation of the evolutionary designs, the current insights from source-term research as described in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," June 1992, regarding fission-product releases into the containment would be utilized.

10 CFR 50.34(f)(2)(xxviii) requires the evaluation of pathways that may lead to control room habitability problems "under accident conditions resulting in a TID 14844 source term release." Similar wording appears in subparagraphs (vii), (viii), and (xxvi). ABB-CE has implemented the new source term technology summarized in Draft NUREG-1465, however, not the old TID 14844 source term cited in the regulation.

The NRC staff has encouraged the development and implementation of the new source term technology and, as stated below, concurs with ABB-CE's approach. Based on the staff's review and ABB-CE's commitments in Chapter 15 of CESSAR-DC, the staff concludes that the special circumstances described in 10 CFR 50.12(a)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because ABB-CE has proposed acceptable alternatives that accomplish the intent of the regulation. On this basis, the staff concludes that an exemption from the requirements of 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) is justified.

In the draft Commission paper on the revised accident source term, the staff identified 12 issues which are applicable to evolutionary and passive ALWR designs.

The issues, which apply to the System 80+ design, are discussed in the sections that follow.

1. Memorandum. James M. Taylor to the Commissioner's, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light Water Reactor Designs." February 10, 1994. [NUDOCS Accession No. 94030269]

15A.2 Truncation of NUREG-1465 Source Term for Use in DBA Assessment

The staff has determined that the appropriate application of the source term expressed in draft NUREG-1465 should be based on the use of gap release and the early in-vessel releases for design-basis-accident (DBA) evaluations. The staff considers the inclusion of the late in-vessel and the ex-vessel source terms to be overly conservative for DBA purposes. In essence, these events are of such low probability that they are not credible in terms of satisfying applicable requirements of 10 CFR Part 100. The late in-vessel and the ex-vessel source terms were used by the staff, however, for the severe-accident consequence assessments in this report.

Although the makeup and timing for the new source term has changed somewhat from that described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962, the staff believes that the new source term provides an increased level of confidence into the actual releases of radioactive materials which may occur.

Table 15A.1 lists the values for PWR releases into the containment for gap releases and early in-vessel releases. Table 15A.2 compares the gap and in-vessel releases in draft NUREG-1465 with TID-14844 source terms for gap and in-vessel releases. As can be seen from the data, the principal differences related to the new source term are mostly related to release timing and the isotopic composition of the release.

15A.3 Iodine Chemical Form

15A.3.1 Transition From Predominantly Elemental Iodine to Particulate Form

RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," specifies that fission-product release into the containment consists of 100 percent of the core inventory of noble gases and 50 percent of iodines (half of which are assumed to deposit on interior surfaces of the containment very rapidly). The chemical form of iodine is specified predominantly elemental iodine (91 percent), with 5 percent assumed to be particulate iodine and the remaining 4 percent assumed to be in the organic form. One percent of "solid" fission products included in TID-14844 was dropped from consideration in RG 1.4.

In draft NUREG-1465, the staff concluded that iodine entering the containment from the reactor core is composed of at least 95-percent cesium iodide (CsI) in particulate form with no more than 5 percent of iodine (I_2) and hydrogen iodide (HI). Once within the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution and will deposit onto the interior surfaces. The staff also stated in NUREG-1465 that the radiation-induced conversion of iodide (I^-) in water into elemental iodine (I_2) is strongly dependent on the pH. Without pH control, the staff indicated that large fractions of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be released into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained at 7 or above, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine. The EPRI URDs for evolutionary and passive plants and all ALWR designs require that the pH of the water in the containment be maintained at or above 7 (alkaline state) for the entire accident duration to minimize the formation of elemental iodine in the containment water, in order to reduce subsequent release of iodine into the containment atmosphere. The staff agrees with this requirement.

15A.3.2 pH Control and the System 80+ Design

In draft NUREG-1465, the staff referenced NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents," April 1992, for the chemical forms of iodine and its subsequent behavior after entering the containment from the RCS. This report pointed out that, among other things, containment water exposed to air will absorb carbon dioxide to form carbonic acid. This would lower the pH slightly, as carbonic acid is a relatively weak acid. In addition, nitric acid can be formed by the irradiation of water and the nitrogen naturally present in air. The report further showed the decrease in pH resulting from these acid additions for an irradiated solution that contained trisodium phosphate with an initial pH of 9.0.

After issuing draft NUREG-1465 in June 1992, the staff issued NUREG/CR-5950, "Iodine Evolution and pH Control" in December 1992. This report pointed out that the most important acids formed in the containment following a DBA will be nitric acid produced by irradiation of water and air, and hydrochloric acid produced by irradiation (radiolysis) and heating of electrical cable insulation (pyrolysis). Electrical cables typically used in operating reactor plants have an ethylene-propylene rubber elastomer as an insulator with a jacket of Hypalon. Hypalon is a chlorosulfonated polyethylene which contains 27 weight percent of chlorine as described by its chemical formula.

In the System 80+ design, borated water with 4000 to 4400 ppm boron in the IRWST will be used for the containment spray solution. This water contains no

Table 15A.1 Release Fractions for Proposed Reactor Accident Source Terms (PWR releases into containment*)

Nuclide	Gap Release**	Early In-Vessel+
Noble Gases	0.05	0.95
Iodine	0.05	0.35
Cesium	0.05	0.25
Tellurium	0	0.15
Strontium	0	0.03
Barium	0	0.04
Ruthenium	0	0.008
Cerium	0	0.01
Lanthanum	0	0.002

*Values shown are fractions of core inventory.
**Duration = 0.5 hour.
+Duration = 1.3 hours.

Table 15A.2 Comparison of gap and in-vessel releases in NUREG-1465 with TID-14844 source terms for PWRs*

Nuclide	TID-14844	NUREG-1465 ABB-CE System 80+
Noble Gases	1.0	1.0
Iodine	0.5	0.4
Cesium	<0.01	0.3
Tellurium	<0.01	0.15
Strontium	<0.01	0.03
Barium	<0.01	0.04
Ruthenium	<0.01	0.008
Cerium	<0.01	0.01
Lanthanum	<0.01	0.002

*Values shown are fractions of core inventory.

chemical additive for pH control during the initial stage of a LOCA. In the CESSAR-DC, ABB-CE states that the pH of the water in the IRWST is raised to a level of 7.0 to control postaccident evolution of elemental iodine and to minimize corrosion of the stainless steel in the containment.

A total mass of 18,930 kg (41,734 lb(mass)) of trisodium phosphate dodecahydrate (purity of 92 percent) is stored in baskets in the IRWST holdup volume. During a LOCA, this volume fills with water and the resulting solution overflows into the IRWST. The baskets (attached to the primary shield wall of the holdup volume) have a solid top and bottom with mesh sides to permit submergence of the trisodium phosphate. The baskets are evaluated above the normal operating water level in the holdup volume and below the IRWST spillway. The configuration of the IRWST spillway piping will promote mixing of the containment spray solution. The staff estimates it will take at least 7.5 hours to achieve a complete mixing and a pH of 7.0 in the containment spray solution based on a three-water volume turnover with a 18,927 L/min (5000-gpm) spray flow rate.

In its evaluation, ABB-CE considers neither nitric acid nor hydrochloric acid formation following a LOCA. The staff evaluated postaccident iodine evolution and pH control, complete with effects of nitric and hydrochloric acids, in NUREG/CR-5950 for the Palo Verde Nuclear Station (a System 80 plant). In that evaluation, the staff calculated a pH of 7.7, assuming the containment sump water at Palo Verde will contain 4400 ppm boron as boric acid plus 2000 ppm as phosphate without considering nitric and hydrochloric acid formation. The staff further calculated that even with hydrochloric acid influx from the approximately 18,000 kg (39,683 lb(mass)) of Hypalon electrical cable insulator used in the Palo Verde containment, the containment sump water will be able to maintain a pH in excess of 7.0. The staff's calculation, however, had the following two limitations:

- It terminated 4 days from the onset of a LOCA.
- It did not consider additional acids that may be produced by pyrolysis.

These limitations notwithstanding, however, the staff believes that most of the hydrochloric acid would be generated from electrical cable insulators within the first 4 days of an accident and the effects of pyrolysis for acid formation compared to that of radiolysis are negligible. The System 80+ design is capable of providing the IRWST water with 4400 ppm boron as boric acid and 2000 ppm as phosphate following a LOCA. Limiting the use of electrical cable insulators (Hypalon or equivalent) to less than 18,000 kg in the System 80+ containment is COL Action Item 15.A.3.2-1

15A.4 Equipment Qualification/Survivability

In evaluating the radiological consequences of analyzed accidents, the staff proposed to define the radiation environment from a design-basis accident to be the environment resulting from releases from the reactor coolant from the pellet-cladding gap (gap activity) and in-vessel releases (see Section 3.11 of this report for equipment qualification for design-basis-accidents). In addition, however, the staff is also defining the radiation environment resulting from severe accidents to include ex-vessel releases and late in-vessel releases. In considering the resultant radiation environment, the staff concludes that for safety-related equipment relied on to cope with severe accidents, there should be a reasonable level of confidence that this equipment will survive the severe reactor accident environment. This area is further discussed in Chapter 19 of this report (see Section 19.2.3.3.6 for equipment survivability under severe accident conditions).

15A.5 Iodine Deposition on Steamlines and Condenser

This does not apply to the System 80+ design.

15A.6 Fission-Product Holdup in Secondary Containment

This does not apply to System 80+ design.

15A.7 Fission-Product Release Timing

Previous staff analyses and regulatory guidelines assumed an instantaneous release of fission-products into the containment where they are assumed to be available immediately for release to the environment.

In draft NUREG-1465, more realistic fission-product release timing mechanisms were assumed. For example, fission-product gap activity releases for a large-break LOCA was estimated to commence no earlier than 10 to 30 seconds for a PWR. Further, in draft NUREG-1465, the staff indicated that early fission-product early in-vessel releases were estimated to start no earlier than 0.5 hour for PWRs. As noted in draft NUREG-1465 (Table 3.6, "Release Phase Durations for PWRs and BWRs"), the duration of the gap activity release considered acceptable by the staff is 0.5 hour and the duration of the early in-vessel failure release phase of the LOCA is 1.3 hours. ABB-CE analyzed the radiological consequences of the design-basis LOCA assuming the timing presented in draft

NUREG-1465. ABB-CE conducted its analysis in accordance with staff guidelines and, therefore, is acceptable.

15A.8 Aerosol Deposition in Containment

To determine radioactive aerosol removal in the System 80+ containment following a LOCA (in unsprayed region), control element assembly (CEA) ejection accident, and feedwater-line break inside containment, ABB-CE used the methodology described in Appendix A, "Physical Processes Associated With Aerosol Removal from the Containment Atmosphere," in an EPRI report titled "Licensing Design Basis Source Term Update for the Evolutionary Advanced Light-Water Reactor," September 1990 (EPRI 1990). The EPRI report references the mechanistic correlation developed by the industry degraded core rulemaking program (IDCOR). The correlation establishes the functional relationship between a dimensionless removal rate constant for sedimentation as a function of dimensionless suspended mass concentration.

ABB-CE's proposed removal rate of 0.15 per hour by sedimentation corresponds to an airborne concentration of approximately 0.02 gm/m^3 in the correlation. It neither considered diffusiophoresis nor hygroscopicity which, by doing so, leads to a more conservative estimate. For the CEA ejection accident, ABB-CE assumed a puff release of approximately 6.8 percent of the gap inventory or approximately 2000 gm of solids (neglecting coolant mist from the blowdown). With a System 80+ containment-free volume of approximately 1 x 10^5 m^3, this amount of solids leads to an airborne concentration of approximately 0.02 gm/m^3 and an aerosol removal rate of 0.15 per hour.

The staff's model for evaluating natural deposition processes in the containment is in its final stages of development (scheduled to be published as NUREG/CR-6189) under a contract with the Sandia National Laboratory. The staff used major insights from that model to perform a comparative analysis with the model used by ABB-CE. The staff's model uses three natural processes for removing radioactive aerosol from the containment atmosphere over the entire period of an accident (30 days): (1) sedimentation mechanism of gravitational settling, including aerosol agglomeration, (2) diffusion mechanism of diffusiophoresis and thermophoresis, and (3) turbulent diffusion to walls. Neither the staff's nor ABB-CE's model explicitly considers hygroscopicity of the aerosol particles except to argue that water adsorption makes particles spherical. The staff's model predicts higher rates of aerosol deposition than does ABB-CE's model during most of the period of fission-product releases and for about 16 hours after fission-product release is complete.

For a duration of 24 hours after fission-product release began, ABB-CE's model predicted more conservative decontamination (i.e., less deposition) of the containment atmosphere by natural aerosol processes than the staff's model. By this time, more than 95 percent of the fission products released to the containment have been deposited as a result of natural aerosol-removal processes. Both models predict rather extensive deposition of the remaining radioactive aerosol over the next few days. On the basis of these comparisons, the staff concludes that ABB-CE's model is adequately conservative and, therefore, it is acceptable.

15A.9 Aerosol Removal by Suppression Pool

This does not apply to System 80+ design.

15A.10 Containment Spray Removal

GDC 41, 42, and 43 of Appendix A to 10 CFR Part 50 require that systems which control fission products reduce the concentration that may be released to the environment. The containment spray system (CSS) reduces containment pressure and temperature and removes airborne radioactive fission products in the containment atmosphere following a LOCA.

The EPRI URD for evolutionary plant designs requires a CSS. ABB-CE has a safety-grade CSS in the System 80+ design. The CSS consists of two redundant and independent trains powered from separate sources independent of offsite power. Each of the two containment spray pumps has a design flow rate of 18,900 l/min (5000 gpm). The two containment spray pumps are automatically started by a safety injection actuation signal (SIAS) and spray borated water (4000 to 4400 ppm as boron) to the containment atmosphere, taking suction from the in-containment refueling water storage tank. The normal operating water volume of this tank is 2.1 x 10^6 L (545,800 gal). The CSS is designed to operate throughout the entire duration of a LOCA.

The total free volume of the System 80+ containment is 9.25 x 10^4 m^3 (3.34 x 10^6 ft^3) of which the effective spray volume is 7.67 x 10^4 (2.74 x 10^6 ft^3) (approximately 82 percent of the containment's free volume). ABB-CE assumes that the remaining 18 percent is unsprayed. ABB-CE also assumes that the average weighted fall height of spray droplets is 25.9 m (84.8 ft). To obtain a weighted average value of the spray removal coefficient for the entire sprayed volume (the sum of Regions I, II, and III), ABB-CE weighted the individual spray regions by the number of nozzles included in each of three sprayed regions. ABB-CE calculated mixing between the sprayed and unsprayed volumes of the containment using the method described in the EPRI evolutionary plant source term paper (EPRI 1990). This method is based on the density increase in the sprayed volume and the resulting density-driven flow exchange with the unsprayed volume as the containment cools due to the effects of spray.

In their application of the revised accident source term to the System 80+ containment spray system, the staff and ABB-CE deviated from the guidance given in RG 1.4 and the review procedures stated provided in SRP Section 6.5.2. The staff considered the removal of airborne fission products in particulate form by spray as a first-order differential of particulate concentration in the containment atmosphere; the particulate removal coefficient is given in a mathematical equation form in the SRP. ABB-CE augmented this equation by incorporating diffusiophoretic deposition due to steam condensation on the dispersed spray droplets. This argumentation is done by using the SWNAUA computer code ("SWNAUA VER02.LEVOO, "Aerosol Behavior in a Condensing Atmosphere - Diffusiophoresis and Spray Version on a PC," NU-185, May 1993) which is a further modification of the NAUA-4 code (Bunz, H. et al., "NAUA Mod 4: A Code for Calculating Aerosol Behavior in LWR Core Melt Accidents, Code Description and User's Manual, Preliminary Description," March 1982) to include the effects of removal by diffusiophoresis. No effects of steam condensation on particules has been included in the analysis of the CE System 80+ containment spray system.

In implementing the revised accident source term for evolutionary reactor designs, the staff approached the removal of airborne fission products in particulate form by spray in an entirely different way from that ABB-CE. The staff developed a mechanistic and simple model that can be used to estimate aerosol removal by sprays without having to use such detailed systems codes as NAUA-4 or CONTAIN. It is described in detail in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," June 1993. The staff developed its model using current knowledge of the physical phenomena involved in spray performance (e.g., observed spray performance data). With this model, the staff conducted a quantitative uncertainty analysis of spray performance using a Monte Carlo method to sample 20 uncertain quantities related to phenomena of spray droplet behavior as well as the initial and boundary conditions expected to be associated with DBAs. Spray water flux into the containment and the fall distances of spray droplets are plant design specifics.

The staff calculated fission-product removal coefficients (lambda values) for the System 80+ containment spray system for best estimate, upper bound (90-percent confidence that lambda values are less than or equal to indicated values), and lower bound (90-percent confidence that lambda values are greater than or equal to indicated values) using the staff's model described in NUREG/CR-5966. The staff evaluated the spray model proposed by ABB-CE and compared it with the model developed by the staff as follows:

(1) The average spray droplet size of 1000 micrometers (μm) used by ABB-CE is more conservative than the distribution of droplet sizes (200 to 1200 μm) used in the staff's model.

(2) ABB-CE's correlation used to calculate terminal velocities for droplets and its capture efficiencies are more conservative than those used in the staff's model.

(3) Both ABB-CE and the staff assume that the radioactive aerosols are not hygroscopic. The staff did not consider the aerosols to be hygroscopic because such hygroscopic components as CsOH and CsI will be greatly diluted by nonhygroscopic materials following a reactor accident.

(4) Particulate capture efficiencies used by ABB-CE are different from those used in the staff's model and, coupled with the conservative terminal velocity correlation, would yield more conservative results compared to particulate capture efficiencies used by the staff.

(5) ABB-CE used the diffusiophoretic capture of aerosols; the staff neglected this.

(6) The staff assumed that sprayed and unsprayed regions in the containment are well mixed; ABB-CE assumed that the sprayed and unsprayed portions of the sprayed region are well-mixed but that mixing between the sprayed and unsprayed regions has a finite rate.

The staff performed a comparative analysis of ABB-CE's spray model with its own model. The staff used the lower bound spray removal coefficient values in its analysis and found that ABB-CE's model produced spray coefficients which were conservative relative to the staff's values. Therefore, the staff finds that ABB-CE's spray model proposed for the System 80+ containment design is acceptable.

15A.11 ESF Filtration/Adsorber Systems

The System 80+ design has provided engineered safety feature (ESF) filtration and adsorber systems where credit was needed in the DBA analysis. ESF-grade systems were provided for the annulus ventilation and containment ventilation purge systems (HEPA filters only), control room ventilation system (HEPA filters and charcoal adsorbers), and the fuel handling building (HEPA filters only). The staff's evaluation of the control room ventilation system appears in Section 6.4 of this report.

The annulus ventilation system consists of two redundant ventilation systems; each system comprises of a fan, a filter train, associated ductwork, dampers, and necessary controls. The annulus ventilation system in the System 80+ design did not take credit for iodine removal by charcoal filtration. Likewise, it took no credit for removal of iodines in either the elemental or organic form and assumed a particulate removal efficiency of 99 percent. In the staff's review of the analysis of the radiological consequences of a design-basis LOCA, credit was given only for the removal of particulate iodines. ABB-CE's analysis also assumed no credit for removal of other iodine forms. Both analyses demonstrated the capability of the System 80+ design with respect to radiological consequences of a LOCA. In evaluating the radiological consequences of a fuel handling accident in either the containment or the fuel building, credit was taken for the operation and filtration of their respective ventilation systems.

15A.12 Atmospheric Dispersion Model for Control Room Habitability Assessment

The staff's analysis appears in Section 6.4 of this report.

15A.13 Failure of Passive Containment Cooling System

This does not apply to System 80+ design.

16 TECHNICAL SPECIFICATIONS

16.1 Introduction

The staff review of the System 80+ technical specifications (TS) was closely coupled to the development of the improved standard technical specifications (STS) under the TS Improvement Program in accordance with the Commission Policy Statement on TS Improvement 58 FR 39132 (SECY-93-067). Since the System 80+ design evolved primarily from the System 80 design, most of the System 80+ TS were modeled after NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants" (January 1991). These improved STS contain the benefits of the accumulated operating experience from currently operating light-water reactors.

The System 80+ design, however, has some major changes from the System 80 design, particularly in the containment, which consists of a much larger free containment volume to account for severe accident concerns. Other specific changes include the reactor cavity and flooding system, the in-containment refueling water storage tank (IRWST), the Nuplex 80+ advanced control complex, the hydrogen control system, the addition of a safety depressurization system and the addition of a combustion turbine generator (CTG). The System 80+ TS address these changes and the issue of shutdown risk considerations. On this basis, the staff's review of the System 80+ TS concentrated on the differences from the System 80 STS (NUREG-1432).

The staff forwarded to ABB-CE, its comments from the System 80+ proof and review of TS, including Amendments U and V, for resolution and incorporation into the final TS. The final TS were produced in the industry format and certified as accurate by ABB-CE in Amendment W to the CESSAR-DC.

16.2 Evaluation

The staff evaluated the System 80+ TS to confirm that they will preserve the validity of the design plant as described in the CESSAR-DC by assuring that System 80+ plants will be operated (1) within the required conditions bounded by the CESSAR-DC, and (2) with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents postulated in the CESSAR-DC.

The System 80+ design includes a large spherical steel primary containment inside a cylindrical concrete shield building that acts as a secondary containment. The 1.52 m (5-ft) annulus between the primary and the secondary containments provides for a means to collect and filter the leakage that might escape the primary containment during postulated design-basis accidents to reduce potential radiation release. The TS include a limiting condition for operation (LCO) and surveillance requirement on the annulus ventilation system.

Another enhancement to the design is the IRWST located inside the large spherical steel containment. Containment water is collected and flows into the tank following large breaks. The IRWST provides borated water for refueling activities as well as for the emergency core cooling systems. Accordingly, TS for reduced inventory are expanded to provide more detailed coverage for operation in Modes 5 and 6.

The reactor cavity and its flooding system are of new design to prevent core debris transport and maximize cooling. The flooding system incorporates passive gravity flow from the IRWST to the cavity via a holdup volume tank (HVT). The TS include surveillance requirements on the operability of the connecting HVT valves.

Hydrogen igniters are provided to meet NRC requirements to accommodate the oxidation of 100 percent of active zircalloy cladding and maintain the global hydrogen concentration in the containment below 10 volume percent. TS include surveillances to test the 80 hydrogen igniters contained in the system during each refueling outage. The hydrogen igniters can be powered from Class 1E power supplies, including the Class 1E batteries. A minimum set of igniters can be powered from the batteries during a station blackout scenario.

The new safety depressurization system rapidly depressurizes the reactor coolant system, when normal processes are not available, to allow an operator to initiate primary system feed and bleed (using the safety injection pumps) to remove decay heat following a total loss-of-feedwater event. Depressurization is accomplished by opening motor-operated valves connected to the pressurizer, which discharge to the IRWST. TS include LCO for the reactor coolant gas vent system and for the rapid depressurization function.

The Nuplex 80+ advanced control complex uses a combination of commercial instrumentation and control (I&C) technologies and incorporates lessons from previous control rooms to make an orderly, natural transition to an all-digital computer-based control room. This approach is reflected in a number of revisions to the I&C TS, surveillance requirements, definitions, and bases.

The System 80+ design has a CTG, in addition to two 1E emergency diesel generators. The staff requested ABB-CE to perform a probabilistic risk sensitivity study for crediting the CTG in the System 80+ design when one of the two 1E emergency diesel generators (EDG) is out of service, with or without an external event. The results indicated a 1-4 percent core damage frequency (CDF) increase, which is considered within the acceptable range of risk. Furthermore, the System 80+ design has the capability of 100 percent load rejection using the main turbine-generator to supply house loads without challenging the EDGs, and its CDF increase is sufficiently low during station blackout and loss-of-offsite power events to justify an EDG outage time of 14 days, with appropriate surveillance requirements.

Considering the recent reactor operating events at Sequoyah and Salem, in which an undetected reactor vessel water level decrease occurred in mode 5, ABB-CE modified the System 80+ TS to require operable reactor vessel level instrumentation during modes 4 and 5 until a transition is made into mode 6. A TS surveillance requirement was also added to perform a channel check of the reactor vessel level instrumentation every 6 hours.

The Idaho National Engineering Laboratory (INEL) conducted an independent audit of the System 80+ TS. TS issues identified by the audit were successfully resolved.

In the DSER, ABB-CE was asked to consider how, and in what priority, added design features and additional margins should be reflected in the TS requirements. This was designated as DSER Open Item 16-2. On the basis of the staff's review of the System 80+ TS applicable to the added design features described above, DSER Open Item 16-2 is resolved.

16.3 Conclusion

On the basis of its review of the System 80+ TS, as discussed above, the staff concludes that the System 80+ TS are consistent with the regulatory guidance contained in the System 80 STS (NUREG-1432) and contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the staff concludes that the System 80+ TS satisfy 10 CFR 50.34 and 10 CFR 50.36, and FSER Confirmatory Item 16-1, as referenced in the advance version of the FSER, is resolved.

17 QUALITY ASSURANCE

17.1 Quality Assurance for Design

17.1.1 General

The quality assurance (QA) program for the design, procurement, and fabrication of the System 80+ plant is described in CESSAR-DC Chapter 17 which references ABB-CE topical report CENPD-210-A, "Quality Assurance Program." The staff based its evaluation of the QA program on its review of the information in the topical report and discussions with ABB-CE's representatives. The staff assessed the QA program to determine if it complies with the requireYments of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the applicable QA-related regulatory guides listed in Chapter 17 of the standard review plan (SRP) (NUREG-0800).

17.1.2 Organization

ABB-CE's Nuclear Systems organization is responsible for the design, procurement, and fabrication of the nuclear steam supply system (NSSS). Nuclear Systems comprises four organizations: Nuclear Systems Engineering, Nuclear Systems Development, Newington Operations, and Electro-Mechanics, each directed by a Vice-President who reports to the President of Nuclear Systems.

QA management is responsible for ensuring that ABB-CE's policy, goals, and objectives are transmitted through levels of management and this is accomplished by distributing QA manuals that contain QA policy statements to these levels of management. It remains the responsibility of functional line management to ensure that the policy, goals, and objectives are met.

Responsibility for nuclear QA rests with the President of Nuclear Systems and is delegated to QA managers who may re-delegate specific activities to other personnel. Such delegation includes authority to stop work for noncompliance with the QA program requirements. Compliance with quality requirements is measured through planned surveillances or audit activities or both, and QA staff follows up with corrective action. QA is independent of other organizations, and QA managers have direct access to the President of Nuclear Systems. ABB-CE's QA organization complies with Appendix B to 10 CFR Part 50 and is, therefore, acceptable.

17.1.3 Quality Assurance Program

Through its topical report CENPD-210-A, Revision 7, "Quality Assurance Program," ABB-CE has adopted a QA program that meets the requirements of Appendix B to 10 CFR Part 50. ABB-CE has committed to comply with the regulatory position of Regulatory Guide (RG) 1.28, (Rev. 3), "Quality Assurance Program Requirements (Design and Construction)," with some exceptions as noted in CENPD-210-A. In addition, ABB-CE has committed to comply with the applicable portions of American National Standards Institute (ANSI), American Society of Mechanical Engineers (ASME) NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," and ASME NQA-2, "Quality Assurance Requirements for Nuclear Facility Application," with an exception as noted in the topical report. However, a review of the QA-related RGs listed in CESSAR-DC Table 1.8-1 indicates that ABB-CE had not revised the table to reflect changes to commitments before the staff issued the draft safety evaluation report (DSER). The DSER open items listed below discuss this in more detail:

(1) CESSAR-DC Table 1.8-1 addresses RGs 1.30, 1.58, 1.64, 1.74, 1.88, 1.123, 1.144, and 1.146. Since these guides (and their referenced standards) have been incorporated into Revision 3 of RG 1.28 (and NQA-1) as committed to in the latest Nuclear Regulatory Commission (NRC)-accepted revision of CENPD-210-A, ABB-CE should have revised Table 1.8-1 to address RG 1.28, Revision 3. This was designated as DSER Open Item 17.1.3-1. In Amendment L, ABB-CE revised Table 1.8-1 to state that RGs 1.30, 1.58, 1.64, 1.74, 1.88, 1.123, 1.144, and 1.146 are superseded by RG 1.28, Revision 3. Therefore, DSER Open Item 17.1.3-1 is resolved.

(2) The latest revision to CENPD-210-A includes a commitment to NQA-2 which has superseded the ANSI standards referenced by RGs 1.30, 1.37, 1.38, 1.39, 1.94, and 1.116. ABB-CE should have revised Table 1.8-1 to reflect this commitment. This was designated as DSER Open Item 17.1.3-2. In Amendment L, ABB-CE revised Table 1.8-1 to state that RG 1.30 was superseded by RG 1.28, Revision 3. Additionally, Note A was inserted with respect to RGs 1.37, 1.38, 1.94, and 1.116. In Amendment N, Table 1.8-1 was revised to insert Note A with respect to RG 1.39. Note A states that the QA program description for System 80+ (CENPD-210-A) commits to NQA-2. Therefore, this DSER open item is resolved.

The QA program applies to all safety-related items and those services engineered, procured, and manufactured by ABB-CE. Highlights of the program are described below.

Procedures require formal training and indoctrination of personnel performing activities affecting quality to ensure they are suitably trained and their proficiency is maintained.

The QA program establishes a system for design control. The system is documented and controlled by procedures and instructions. These procedures and instructions describe the responsibilities and links between each organizational unit that has an assigned responsibility. Distribution lists and master lists of project drawings and specifications are maintained to ensure timely and accurate access to latest applicable documents.

ABB-CE has established and documented measures for the preparation, review, approval, and control of procurement documents. These measures offer assurance that the procurement documents include or reference regulatory requirements, design bases, and quality requirements.

The QA staff will review and approve purchase specifications before they are issued. Review of procurement documents by qualified engineers and QA personnel will ensure that quality requirements are complete and correctly stated. The reviews will also ensure that the quality requirements will be controlled by the supplier and will be verified by QA personnel.

ABB-CE requires its suppliers to identify and control materials, and the QA staff inspects the marking of items before those items are shipped. Material identification and control is assured by requiring a written procedure that is reviewed by the QA group.

ABB-CE requires that in-process and final inspections be performed in accordance with procedures submitted to and found acceptable by ABB-CE. Procedures require that inspection personnel be qualified and that records of qualification be maintained. These procedures require that inspection personnel be organizationally independent from personnel who perform the work being inspected.

Suppliers must maintain a system that identifies, documents, and controls nonconforming items to prevent their inadvertent use. QA staff reviews and approves nonconformance actions. The engineering group evaluates and dispositions nonconformances, and the QA staff reviews these actions. QA personnel also verify proper corrective action. ABB-CE writes nonconformance reports regarding the affected item and forwards them to the utility for handling.

ABB-CE executes a comprehensive system of planned and documented audits to verify product quality and compliance with the QA program. The audits use preestablished checklists, ensuring compliance with all aspects of 10 CFR Part 50, Appendix B, including the quality-related aspects of design, procurement, manufacture, storage, shipment, and reactor site activities. The QA program requires that suppliers, too, audit their operations and their subvendor's operations to verify conformance with quality requirements. The audits include quality-related practices, procedures, instructions, and conformance with the QA program. The QA group audits the suppliers and selected subvendors. Written reports are forwarded to managers of the area audited and to corporate management. Followup audits ensure corrective action.

The staff also reviewed the list of items to which the QA program applies (CESSAR-DC Table 3.2-1). The list of items was reviewed by each technical review branch in the NRC to ensure that the safety-related items within the scope of each branch's review are under the QA program controls. The staff had questions in this area; these are listed below:

ABB-CE revised Sheet 1 of CESSAR-DC Table 3-2.1 to include "Core Support Structures and Internal Structures Important to Safety." These items are shown as Safety Class 1, seismic Category I, and are subject to the QA requirements of 10 CFR Part 50, Appendix B. The addition of the words "important to safety" implies that there are some core support structures and internal structures that are not important to safety and are, therefore, not subject to the QA requirements of 10 CFR Part 50, Appendix B. ABB-CE should identify any such items. This was designated as DSER Open Item 17.1.4-1. In Amendment N, ABB-CE revised Table 3-2.1. The revised table deleted the statement "important to safety" and stated, in Footnote 7, that the support structures and internal structures are designed to the criteria described in Section 3.9.4.3. Therefore, DSER Open Item 17.1.4-1 is resolved.

ABB-CE revised CESSAR-DC Table 3.2-1 to show three items as seismic Category I with the QA requirements of 10 CFR Part 50, Appendix B, not applicable. These are the spent fuel racks (Sheet 3), the new fuel racks (Sheet 3), and the hydrogen igniters (Sheet 13). All other seismic Category I items in the table are shown as having the QA requirements of 10 CFR Part 50, Appendix B, applicable. ABB-CE should have justified the exclusion of the spent fuel racks, new fuel racks, and hydrogen igniters. This was designated as DSER Open Item 17.1.4-2. ABB-CE revised Table 3.2-1 in Amendment N to resolve this issue. The spent fuel and new fuel racks have been changed to Quality Class 1 for which the requirements of 10 CFR Part 50, Appendix B apply. ABB-CE also, in Amendment N, revised Table 3.2-1 requirements to show the hydrogen igniters as having a safety class of NSSS and a Quality Class of 2. The Quality Class 2 (intermediate-level quality class) is acceptable for the hydrogen igniters. See DSER

Open Item 17.1.4-3 (below) with respect to Quality Classes. Therefore, DSER Open Item 17.1.4-2 is resolved.

As noted in the response to the request for additional information (RAI) Q260.24, ABB-CE revised CESSAR-DC Table 3.2-1 to specify QA requirements as either "Q" (Appendix B applies) or "N" (Appendix B does not apply). The table no longer shows Quality Class 1 or 2, and this tends to negate the acceptability of the response to Q210.7 and Q260.24.c. The CESSAR-DC needed to clearly show that the pertinent provisions of CENPD-210A (or some other described QA program) are applied to the items in the table with QA requirements shown as "N." ABB-CE should commit to use a graded approach that bases the QA requirements on the specific functions and their importance to safety. The items should have included those specified in Section 3.3.1.4 of ANSI/American Nuclear Society (ANS)-51.1 (including the safety parameter display system or its equivalent), fire protection, non-safety-related "anticipated transient without scram (ATWS) items" specified in 10 CFR 50.62, and non-safety-related items specified in 10 CFR 50.65. This was designated as DSER Open Item 17.1.4-3. In Amendment N, ABB-CE outlined a graded approach to QA by describing three quality classes, QC-1, QC-2, and QC-3. QC-1 is the highest level and meets the requirements of Appendix B to 10 CFR Part 50. QC-2 is an intermediate class, and QC-3 is for items not classified as QC-1 or QC-2. All items in Table 3.2-1 have been classified as QC-1, QC-2, or QC-3. Therefore, DSER Open Item 17.1.4-3 is resolved.

In the response to RAI Q260.26, ABB-CE states that no special QA program requirements are necessary for control grade ATWS equipment. However, Generic Letter 85-06 gives explicit QA guidance required by 10 CFR 50.62. ABB-CE should have included a commitment in the CESSAR-DC to meet this guidance or should have described some other way of meeting the regulation. The term, "control grade equipment" was also used in the response to Q440.110. ABB-CE should have clarified what equipment is "control grade" and what QA standard is applied to it. If this information were already in the CESSAR-DC, a specific reference to it would be acceptable. This was designated as DSER Open Item 17.1.4-4. In Amendment R, ABB-CE revised Table 3.2-1 to require a Quality Class 2 (see above) for all equipment required to comply with 10 CFR 50.62 for ATWS equipment. Therefore, DSER Open Item 17.1.4-4 is resolved.

The response to RAI Q260.7 begins: "As described in the response to RAI 270.1, ABB-Combustion Engineering believes that all structures, systems, and components (SSCs) which are not safety-related (i.e., non-nuclear safety) are covered via ANSI 51.1." The staff reviewed both the March 15, 1991, and the February 12, 1992, responses to RAI Q270.1, and found no reference to the ANSI-51.1 standard. ABB-CE should have clarified this reference. This was designated as DSER Open Item 17.1.4-5. In Amendment R, ABB-CE revised Table 3.2-1 to require a graded approach to QA for all equipment specified in Section 3.3.1.4 of ANSI/ANS-51.1. Section 3.3.1.4 of ANSI/ANS 51.1 applies to non-nuclear safety equipment. Therefore, DSER Open Item 17.1.4-5 is resolved.

The response to RAI Q260.27 indicates that all SSCs that are not safety-related are covered via ANSI 51.1. The staff disagreed. For example, ANSI/ANS-51.1 does not address fire protection or the non-safety-related items specified in 10 CFR 50.62. Nor does ANSI/ANS-51.1 describe what QA controls are required, specifying only that selected requirements from ANSI/ASME NQA-1 shall be applied. ABB-CE should have revised its response to RAI Q260.27. This was designated as DSER Open Item 17.1.4-6. In Amendment R, ABB-CE revised Table 3.2-1 to require a graded approach to QA for all equipment specified in Section 3.3.1.4 of ANSI/ANS 51.1 and equipment required to comply with 10 CFR 50.62. (See also the resolution to DSER Open Item 17.1.4-3 above). Therefore, DSER Open Item 17.1.4-6 is resolved.

In regard to Three Mile Island-2 (TMI-2) Item I.F.2, ABB-CE should have included that part of the response to RAI Q260.28 that states: "One portion of subpart 3 (inclusion of QA personnel in design activities) is covered" in the CESSAR-DC. This was designated as DSER Open Item 17.1.4-7. In Amendment L, ABB-CE revised Section 17.1 of CESSAR-DC to state that the inclusion of QA personnel in the design activities is covered by the QA program for the System 80+ design, described in CENP-210-A. Therefore, DSER Open Item 17.1.4-7 is resolved.

The staff performed a QA design control implementation inspection at ABB-CE's offices during the week of February 14, 1994 (NRC Inspection Report 99900401/94-01). The purpose of the inspection was to examine the effectiveness of the design control implementation for work involving the System 80+ design. The inspection was focused on design activities associated with CESSAR-DC Chapters 6 and 15 related to accident analysis. While the inspection confirmed the integrity of ABB-CE's design control implementation, the team identified that ABB-CE had made the decision to not require independent design control provisions until the detailed design is developed (i.e., post-FDA). This decision was originally delineated in internal ABB-CE correspondence D-NE-87-031 (dated June 26, 1987) and further documented in the ABB-CENP System 80+ QA Plan, 18386-Q0-001. In particular, ABB-CE design control procedures QPI 0304, "Design Analysis" and QPI 0306, "Design Verification" have been deferred. These procedures govern the methodology to document design analyses and the performance of design verification activities. The providers of engineering services (i.e., Stone and Webster Engineering Corporation, Duke Engineering Services, Inc., and ABB-Impell) to ABB-CE have performed independent design verification on portions of their design.

The design verification methods of QPI 0306 require verification of the appropriateness of design assumptions, input data utilized, and correctness of analysis methods by a qualified independent reviewer. Various options for performing the independent design verification include: performance of design reviews; use of alternate calculations to verify the design; or the conduct of qualification tests (or a combination of the three methods). ABB-CE did have a supervisor of the calculation preparer perform an engineering review for overall reasonableness of the calculation. In addition, ABB-CE has performed three integrated reviews of CESSAR-DC information for technical and editorial consistency. ABB-CE plans to perform the complete independent design verification at a later point in time, but prior to initial criticality.

The staff expectations during the course of the Chapter 17 QA review were that ABB-CE would fully implement the provisions of the design control program as described within ABB-CE Topical QA Report, CENPD-210A, revision 7A as committed to in CESSAR-DC. The QA Topical Report further commits to RG 1.28 (Rev. 3) and NQA-1 as the approach to implement timely independent design verification to the degree that it is technically feasible to perform. The staff acknowledges that there are technical bases for deferring the conduct of portions of design verification such as when it is necessary to obtain as-procured equipment characteristics, results from testing of plant equipment, and results from plant as-built verifications.

While ABB-CE had not conducted the formal independent design verification to date, the staff has concluded that a level of reasonable assurance for the integrity of the System 80+ design process has been obtained through: (1) ABB-CE conducting supervisory reviews of calculations; (2) the performance of three ABB-CE integrated reviews of CESSAR-DC information for consistency; (3) NRC staff reviews of calculations associated with the accident analysis and structural aspects of the design; (4) the fact that the computer codes utilized for the accident analysis have been previously design verified and any code changes have received line-by-line verification; (5) the fact that System 80+ is an evolutionary design that takes advantage of previously performed design work that has been verified for System 80 designs and overseas designs which have undergone complete design verification; and (6) the positive results of the QA implementation inspection with respect to technical quality of the design documentation that was examined.

The staff requested ABB-CE to formally describe their approach towards performing independent design verification. Pending receipt, evaluation, and confirmation that the ABB-CE design control practices afforded an acceptable level of assurance of the design integrity, this issue was designated as FSER Confirmatory Item 17.1-1 in the advance version of this report.

The staff subsequently met with ABB-CE on March 21, 1994 to discuss the issue of independent design verification of safety analysis calculations that support the CESSAR-DC. As described during the meeting, and further documented in a letter dated April 26, 1994, ABB-CE described its proposed design verification process for design bases events. ABB-CE committed to perform design verification for non-repetitive safety analyses, e.g. those analyses that are not intended to be repeated by a COL applicant. Specifically, the non-repetitive safety analyses include (1) all the design basis event analyses presented in CESSAR-DC Chapters 5, 6, and 15, (2) analyses that set safety-related design parameters, including those described in the Certified Design Material, and (3) a CESSAR-DC Appendix 6B analysis performed to verify the System 80+ capability to safely handle a hypothetical small-break LOCA boron dilution event. ABB-CE will perform a level of design verification consistent with the safety significance of the design analysis.

By letter dated June 16, 1994 (LD-94-041), ABB-CE indicated their completion of the above commitments. Also, ABB-CE supplied information on design verification in Chapter 17 of the CESSAR-DC. This is acceptable. On this basis, FSER Confirmatory Item 17.1-1 is resolved.

17.1.4 Conclusions About the Quality Assurance Program

The staff has reviewed ABB-CE's QA program description for design certification, and the staff has established and verified that all applicable requirements of Appendix B to 10 CFR Part 50 except for independent design verification and design documentation are included in the QA program. Further, the staff has verified that the QA organizations are structured so that ABB-CE can effectively carry out its responsibilities related to quality without undue influence from other groups.

On the basis of its detailed review and evaluation of the QA program described in CESSAR-DC, and by reference to CENPD-210-A, the staff concludes that

(1) The QA organizations within the corporate organization are sufficiently independent from cost and schedule (when opposed to safety considerations) authority to effectively carry out the QA programs, and the QA personnel have access to management at a level necessary to perform the QA functions.

(2) ABB-CE's QA program describes adequate QA requirements and controls which, when properly implemented, comply with the criteria of Appendix B to 10 CFR Part 50 except for provisions relating to independent design verification and design documentation.

Accordingly, the staff concludes that the fundamental requirements for an acceptable design QA program are in place.

17.2 Quality Assurance for Operation

QA programs for construction and operations are beyond the scopes of ABB-CE's application for final design approval and design certification of the System 80+ standard plant design; they are, therefore, combined license (COL) Action Items 17.1-1 and 17.2-1, respectively. Through CESSAR-DC Section 17.1, ABB-CE acknowledges that the COL applicant/holder is responsible for the preparation and implementation of the construction QA program. ABB-CE acknowledges in CESSAR-DC Section 17.2 that the COL applicant/holder is responsible for the operational QA program. For a discussion on the relationship of the COL applicant/holder's operational reliability assurance process to the QA program, see Section 17.3 of this report.

17.3 Reliability Assurance Program During Design Phase

Introduction

In CESSAR-DC Section 17.3, ABB-CE describes the reliability assurance program (RAP) for the design phase of the System 80+ design. ABB-CE performed the design RAP (D-RAP) for its scope of design during detailed design and specific equipment selection phases to assure that important System 80+ reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout plant life. It will identify relevant aspects of plant operation, maintenance, and performance monitoring of important SSCs for the COL applicant's consideration in assuring safety of equipment, preventing loss of critical function, and limiting risk to the public. The COL applicant referencing the System 80+ design will complete the D-RAP for its scope of design and equipment selection. Additionally, the COL applicant should develop and implement a process for risk-significant SSCs, whose objectives are to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits (COL Action Item 17.3.9-1). This process could be described as an operational reliability assurance process (O-RAP) that should be included under existing programs for quality assurance and maintenance. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numeric values used in the PRA through performance monitoring.

In response to an NRC RAI dated October 10, 1991, ABB-CE submitted (by letter dated January 31, 1992) its RAP plan for a System 80+ standard plant. The staff evaluated that submittal. On October 1, 1992, the NRC forwarded to ABB-CE the staff's DSER of the CESSAR-DC for design certification of the System 80+ standard plant that included Section 17.3. In response to the staff's DSER, ABB-CE completely revised the D-RAP. The revised D-RAP was forwarded to the NRC by letter dated December 23, 1992 and was subsequently revised by letters dated January 18, 1993 (Rev. 2) and March 2, 1993 (Rev. 3). These changes were incorporated into Amendment N of the CESSAR-DC, dated April 1, 1993. Further clarification was incorporated in Amendment R, dated August 31, 1993.

Background

The NRC noted the need for a safety-oriented reliability effort for the nuclear industry in the TMI Action Plan (NUREG-0660) Item II.C.4. Subsequently, the NRC began to research the area of reliability assurance began in the early 1980s. This research showed that an operational reliability program based on a feedback process of monitoring performance, identifying problems, taking corrective action, and verifying the effectiveness of these actions was needed and that other NRC initiatives (e.g., maintenance inspections, performance indicators, aging programs, and technical specification (TS) improvements) would address this need. The NRC concluded from this research that an operational reliability program could be implemented most effectively in a performance-based, non-prescriptive regulation, where NRC mandates the level of safety performance to be achieved. For example, licensees could be required to set availability and reliability targets for selected systems and to measure performance compared to the targets.

The TMI item was closed out for operating reactors in October 1988, without further action because several NRC initiatives had effectively subsumed the operational reliability program effort. These initiatives included efforts to (1) improve maintenance and better manage the effects of aging, (2) improve TS, (3) develop and use plant performance indicators, and (4) develop an operational reliability program as an acceptable means of meeting the station blackout rule (10 CFR 50.63).

NUREG-1070, "NRC Policy on Future Reactor Designs," dated March 1985, included the concept of a systems reliability program to ensure that the reliability of components and systems important to safety would remain at a sufficient level. To ensure that reliability objectives are met and to prevent degradation of reliability during operation, the NRC envisioned that the PRA, performed at the design stage, would be used as a tool in making detailed design decisions affecting procurement, testing, and the formulation of operations and maintenance procedures.

In a few specific instances, the NRC is studying or has established reliability targets for systems and components. For example, SRP Section 10.4.9 requires that an acceptable design for auxiliary feedwater system should have an unreliability in the range of 10^{-4} to 10^{-5} per demand. The resolution of Generic Issue B-56 involves efforts to determine, monitor, and maintain the reliability levels for the emergency diesel generators. Additional regulatory bases for key elements of a RAP can be found in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.65.

In SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," dated January 19, 1989, the staff identified several issues for next-generation light water reactors that may go beyond present acceptance criteria defined in the SRP. RAP, as one of these issues, was defined as a program to ensure that the design reliability of safety significant SSCs is maintained over the life of a plant. In SECY-89-013, the staff informed the Commission that RAP would be required for a final design approval or a design certification (FDA or DC). In November 1989, potential applicants for design certification were informed by letter that "the NRC staff was

considering matters that went beyond the current SRP [Standard Review Plan (NUREG-0800)]... that [the NRC] expects these advanced reactor designs to embody." Reliability assurance was identified as one of these matters.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff gave the Commission its interim position that a high-level commitment to a RAP should be required as a generic Tier 1 requirement with no associated inspections, tests, analyses, and acceptance criteria. The details of the D-RAP, including the conceptual framework, program structure, and essential elements, should be given in the CESSAR-DC. The description of the D-RAP in the CESSAR-DC should also (1) identify and prioritize a list of risk-significant SSCs based on the design certification PRA and other sources, (2) ensure that the vendor's design organization determines that significant design assumptions, such as equipment that satisfies the design reliability and unavailability, are realistic and achievable, (3) provide input to the procurement process for obtaining equipment that satisfies the design reliability assumptions, and (4) provide these design assumptions as input to the COL applicant for consideration in the O-RAP. A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process. This is COL Action Item 17.3.1-1.

The RAP consists of two distinct parts: (1) D-RAP and (2) O-RAP. D-RAP involves a top-level program at the design stage that defines the scope, conceptual framework, and essential elements of an effective RAP. D-RAP also implements those aspects of the program that are applicable to the design process. In addition, D-RAP identifies the relevant aspects of plant operation, maintenance, and performance monitoring for the risk-significant SSCs for the operator's consideration in developing an O-RAP. The O-RAP objectives should be incorporated into existing programs (i.e., maintenance and quality assurance) that will be used to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numeric values used in the PRA through performance monitoring.

The staff's final position on RAP was presented in the Commission Paper on the Regulatory Treatment of Non-

Safety Systems (RTNSS), SECY-94-084, dated March 28, 1994. The Commission approved the following applicable regulation for D-RAP.

An application for design certification or a combined license must contain:

(1) the description of the reliability assurance program used during the design that includes, scope, purpose, and objectives;

(2) the process used to evaluate and prioritize the structures, systems and components in the design, based on their degree of risk significance;

(3) a list of the structures, systems, and components designated as risk significant; and

(4) for those structures, systems, and components designated as risk significant:

  (i) a process to determine dominant failure modes that considered industry experience, analytical models, and applicable requirements; and

  (ii) key assumptions and risk insights from probabilistic, deterministic, or other methods that considered operations, maintenance, and monitoring activities.

Each licensee that references the System 80+ design must implement the design reliability assurance program approved by the NRC.

The staff evaluated the CESSAR-DC based on the applicable regulation stated above. The staff limited its review of CESSAR-DC Chapter 17.3 to (1) the development and implementation of the System 80+ D-RAP within the scope of design certification; (2) the development of the System 80+ D-RAP to be used in a combined license (COL) application; and (3) the inclusion of System 80+ information necessary for a COL applicant to develop an O-RAP which should be incorporated under existing programs for quality assurance and maintenance.

The COL applicant would augment the ABB-CE's RAP to reflect plant-specific information and implement those elements applicable during the construction and operation phases. The staff's COL application review will be similar to the design certification review and include an evaluation of the updated (site-specific) PRA, probabilistic, deterministic and other insights (e.g., operating experience) to assess any changes to risk-significant SSCs and site-specific vulnerabilities. The staff will also review the COL applicant's proposed design reliability assurance program plan to determine if it satisfies the above requirements at the time of the COL application. A licensee's RAP plan and implementation will continue to be reviewed throughout the duration of the license to assure conformance with the NRC approved D-RAP.

Evaluation

By letter dated October 10, 1991, the staff stated that the ABB-CE D-RAP submittal should (1) describe the basic framework of a RAP including the scope, purpose, objective, basic definitions, and elements (RAI Question (Q)1); (2) include a discussion on performance goals/targets, problem prediction and recognition, problem prioritization and correction, and problem closeout, when describing the RAP concepts and elements (RAI Q2); (3) describe how RAP will address plant aging concerns (RAI Q3); (4) describe the organizational and administrative aspects for implementing an effective RAP (RAI Q4); (5) describe the approach for providing feedback to the designer when actual plant performance data consistently differs from the ABB-CE's PRA/RAP assumptions (RAI Q5); (6) describe the major programmatic interface between the RAP and such areas as design, construction, startup testing, operations, maintenance, engineering, safety, licensing, QA, and procurement (RAI Q6); (7) provide an example of how the CE RAP would function throughout plant life using a specific SSC identified as risk significant in the PRA (RAI Q7); and (8) describe how the CE RAP differs from the description of a RAP in the Electric Power Research Institute (EPRI) Requirements Document (RAI Q8). These questions included the use of the term reliability assurance program (RAP), however, the intent was for the questions to apply to that portion of the RAP that ABB-CE is responsible for preparing and implementing (e.g., the System 80+ design RAP). The RAI questions served as an outline of the information needed to meet the applicable regulation for D-RAP, as stated above, and explicitly stated the details of this requirement.

By letter dated January 31, 1992, ABB-CE responded to RAI Q1 to Q6 and RAI Q8 and stated that the example to answer RAI Q7 would be provided in a future update. The staff found several deficiencies with that ABB-CE submittal. In the DSER, the staff stated that the ABB-CE RAP should: (1) provide information that identifies risk-significant SSCs, ensures that the plant design provides SSCs at least as reliable as that assumed in the PRA, and maintains SSC reliability levels over the life of the plant (DSER Open Item 17.3.1.1-1); (2) clearly define the scope and objective of a RAP, state the basic definitions, and discuss SSC selection criteria (DSER Open Item 17.3.1.2-1); (3) clarify the control of PRA design assumptions for the RAP, and provide a method to identify and prioritize risk-significant SSCs (DSER Open Item 17.3.2-1); (4) clarify how potential conflicts between the goals of a reliability, availability, maintainability, and inspectability (RAMI) program and PRA will be resolved (DSER Open Item 17.3.3-1); (5) clarify the intent of RAMI (i.e., safety or economics) and explicitly state the priority of safety requirements (DSER Open Item 17.3.3.1-1); (6) clarify the use of the nuclear plant reliability data system or other data in establishing a data base (DSER Open Item 17.3.3.2-1); (7) clarify how the corrective action program (CAP) will verify that equipment is meeting its reliability requirements and is an integral part of the entire reliability program; determine appropriate corrective actions; verify corrective actions have been taken; and feed this information into the data base (DSER Open Item 17.3.3.3-1); (8) clarify which organization is responsible for each reliability centered maintenance (RCM) phase (DSER Open Item 17.3.4-1); (9) submit a better description of the RCM program or the RCM Program Guide (DSER Open Item 17.3.4-2); (10) clarify ABB-CE's intent regarding consistency among the PRA, plant procedures, and TS (DSER Open Item 17.3.5-1); (11) discuss the organizational and administrative aspects of a D-RAP and discuss organizational accountability for implementing the design portion of the RAP (DSER Open Item 17.3.6-1); (12) submit an example of how the RAP would function throughout plant life using a specific system, structure, or component that was identified as risk significant in the PRA (DSER Open Item 17.3.7-1); and (13) discuss in detail how ABB-CE's RAP differs from the EPRI Utility Requirements Document for evolutionary ALWRs, including the rationale for the differences, if any (DSER Open Item 17.3.7-2).

In response to the DSER open items, ABB-CE completely revised the D-RAP and also made minor editorial changes. As stated, this evaluation documents the results of the staff's review of CESSAR-DC Section 17.3. To ensure all RAI and DSER issues are addressed, both the RAI and DSER open items are discussed herein. The changes made by ABB-CE to CESSAR-DC Section 17.3 satisfactorily answered all eight RAI questions. Specifically, the staff determined that: (1) RAI Q1 was answered in Sections 17.3.2, 17.3.3, 17.3.4, 17.3.6, 17.3.7, and 17.3.8; (2) RAI Q2 was answered in Sections 17.3.6, 17.3.9, and 17.3.10; (3) RAI Q3 was answered in Section 17.3.10; (4) RAI Q4 was answered in Section 17.3.5; (5) RAI Q5 was answered in Section 17.3.10; (6) RAI Q6 was answered in Section 17.3.10; (7) RAI Q7 was answered in Section 17.3.11, and (8) RAI Q8 was answered in a letter from ABB-CE (Letter Number (LD)-93-005) dated January 18, 1993. ABB-CE's answers to the RAI questions contained in CESSAR-DC Section 17.3 meets the staff's expectations on RAP for evolutionary ALWR designs explicitly. The details of the staff's evaluation are presented in Sections 17.3.1 through 17.3.11 below.

Seven DSER open items were resolved by the changes to CESSAR-DC Section 17.3 and through other correspondence with ABB-CE. The remaining six of the DSER open items were resolved on the bases that they are no longer applicable due to the changes made by ABB-CE to CESSAR-DC Section 17.3. Specifically, the staff determined that (1) DSER Open Item 17.3.1.1-1 was resolved in Section 17.3.3; (2) DSER Open Item 17.3.1.2-1 was resolved in Sections 17.3.2, 17.3.4, 17.3.6, 17.3.7, and 17.3.8; (3) DSER Open Item 17.3.2-1 was resolved in Sections 17.3.6 and 17.3.7; (4) DSER Open Item 17.3.3-1 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (5) DSER Open Item 17.3.3.1-1 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (6) DSER Open Item 17.3.3.2-1 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (7) DSER Open Item 17.3.3.3-1 was resolved in Section 17.3.10; (8) DSER Open Item 17.3.4-1 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (9) DSER Open Item 17.3.4-2 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (10) DSER Open Item 17.3.5-1 is not applicable to the current revision of CESSAR-DC Section 17.3 and is considered resolved; (11) DSER Open Item 17.3.6-1 was resolved in Section 17.3.5; (12) Open Item 17.3.7-1 was resolved in Section 17.3.11; and (13) DSER Open Item 17.3.7-2 was resolved in a letter from ABB-CE (LD-93-005) dated January 18, 1993. The details of the staff's evaluation are presented in Sections 17.3.1 through 17.3.11 below.

17.3.1 General

In CESSAR-DC Section 17.3.1, ABB-CE states that the System 80+ D-RAP is a program that will be performed by the designers during the detailed design stage and specific equipment specification phases to ensure that the important System 80+ reliability assumptions of the PRA will be considered throughout plant life. The PRA evaluates the plant response to initiating events to substantiate, in part, that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant SSCs. The COL applicant will complete the site specific D-RAP and will have an operations reliability assurance process. The COL applicant/holder should incorporate the operations assurance process objectives into existing programs (e.g. quality assurance or maintenance) that will monitor equipment performance to provide reasonable assurance that the plant is operated and maintained with an acceptably low risk commensurate with PRA assumptions.

ABB-CE states in the CESSAR-DC that the D-RAP will include the design evaluation of System 80+. It will identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for a COL applicant's consideration in assuring safety of the equipment, maintenance of critical functions, and limiting risk to the public. The COL applicant will

specify the policy and implementation procedures for using D-RAP information provided by ABB-CE. This is COL Action Item 17.3.1-1.

In CESSAR-DC Section 17.3.11, ABB-CE describes example of how the D-RAP will be implemented for the component cooling water system (CCWS). The CCWS example shows how the principles of D-RAP will be applied to other systems identified by means of the PRA as significant with respect to risk.

The staff concludes that CESSAR-DC Section 17.3.1 meets the requirement of the applicable regulation for D-RAP to provide a description of the RAP used during the initial design as discussed above in Section 17.3 of this report, and is acceptable.

17.3.2 Scope

In CESSAR-DC Section 17.3.2, ABB-CE states that the scope of the D-RAP includes the design evaluation of the System 80+ and identifies relevant aspects of plant operation, maintenance, and performance monitoring of plant risk-significant SSCs. The PRA for System 80+ and other industry sources will be used to identify and prioritize those SSCs that are important for preventing or mitigating plant transients or other events that could present a risk to the public.

The staff reviewed CESSAR-DC Section 17.3.2 with respect to the scope of the System 80+ D-RAP and concludes that it is responsive to the portion of RAI Q1 and DSER Open Item 17.3.1.2-1 associated with the D-RAP scope, meets the requirement of the applicable regulation for D-RAP to include the scope of RAP, as discussed above in Section 17.3 of this report, and is acceptable.

17.3.3 Purpose

In CESSAR-DC Section 17.3.3, ABB-CE states that the purpose of the D-RAP is to assure that the plant safety, as estimated by the PRA, is maintained as the detailed design evolves through the implementation and procurement phases. Additionally, ABB-CE states that pertinent information is provided in the design documentation to the COL applicant for use so that equipment reliability and availability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

The staff reviewed CESSAR-DC Section 17.3.3 with respect to the purpose of the System 80+ D-RAP and concludes that it is responsive to the portion of RAI Q1 and DSER Open Item 17.3.1.1-1 associated with the D-RAP purpose, meets the requirement of the applicable regulation for D-RAP to include the purpose of RAP as described in Section 17.3 of this report, and is acceptable.

17.3.4 Objective

In CESSAR-DC Section 17.3.4, ABB-CE states that the objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to assure that, during the implementation phase, the plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP also will identify key assumptions regarding any operation, maintenance, and monitoring activities that the COL applicant should consider in developing its O-RAP to assure that such SSCs can be expected to operate with reliability commensurate with that assumed in the PRA. A major factor in the D-RAP is risk-focused maintenance. Maintenance resources are focused on those SSCs that enable the System 80+ risk-significant systems to fulfill their safety-related functions and maintain the safety margins. Also, maintenance is focused on SSCs whose failure may directly initiate challenges to risk-significant systems. All plant modes are considered, including equipment directly relied on in the emergency operating procedures.

The staff reviewed CESSAR-DC Section 17.3.4 with respect to the objective of the System 80+ D-RAP and concludes that it is responsive to the portion of RAI Q1 and DSER Open Item 17.3.1.2-1 associated with the D-RAP objective, meets the requirement of the applicable regulation for D-RAP to include the objective of RAP as described in Section 17.3 of this report, and is acceptable.

17.3.5 ABB-CE Organization for D-RAP

In CESSAR-DC Section 17.3.5, ABB-CE describes the relevant portion of the project organization used for the detailed design of System 80+, as shown in CESSAR-DC Figure 17.3-1. The project organization was integrated and the responsibility to meet the D-RAP objectives rested with the project director. Regular meetings were scheduled to coordinate all the design and D-RAP activities with participation of the engineering manager, PRA and D-RAP program manager, project integration manager, QA manager, regulatory conformance manager, and other managers as necessary. During these meetings, design changes and the impact on the overall plant performance were identified, and discussions about the impact of these changes on plant risk were held. Management meetings were also held in which programmatic issues affecting the System 80+ design were discussed.

Responsibilities for each organization in the D-RAP were assigned. The Project Director was responsible for the programmatic aspects of the plant design as well as the overall direction of the project, design certification issues, and licensing issues. The NSSS design manager was responsible for the design of the NSSS. The regulatory conformance manager had the responsibility of addressing any regulatory concerns and bringing these concerns to the attention of the project integration manager and PRA and D-RAP program manager.

The NSSS design engineering organization was the core of the ABB-CE organization for RAP and it was responsible for the design of the System 80+ NSSS. This group developed the NSSS design and drawings with inputs from the mechanical, instrumentation and control, reactor, and fluid systems subgroups. This organization also developed the PRA models, TS, and some emergency operations guidelines (EOGs).

The PRA and D-RAP program manager was responsible for managing and integrating the D-RAP program and had direct access to the System 80+ project integration manager. The PRA and D-RAP program manager was also responsible for keeping the project integration manager abreast of D-RAP critical items, program needs, and program status. The PRA and D-RAP program manager had the organizational freedom to identify D-RAP problems; he could initiate, recommend, or provide solutions to problems through designated organizations, verify implementation of solutions, and function as an integral part of the design team and final design process.

The PRA and D-RAP program manager was in the ABB-CE's reliability analysis services department which performed reliability analyses, risk assessments, and PRA. This group reported to the NSSS design manager, through the PRA and D-RAP program manager (CESSAR-DC Figure 17.3-1). The PRA input to the D-RAP and any of the System 80+ reliability analyses were performed in this organization and were integrated into the System 80+ design.

The staff reviewed CESSAR-DC Section 17.3.5 with respect to the organizational structure and accountability for implementing D-RAP in the System 80+ design process. The staff concludes that CESSAR-DC Section 17.3.5 was responsive to RAI Q4 and DSER Open Item 17.3.6-1 associated with the organizational structure and accountability for implementing D-RAP in the design process, satisfies the staff position for the D-RAP to ensure that the vendor's design organization determines that significant design assumptions are realistic and achievable as described in SECY-93-087 and as discussed in Section 17.3 of this report, and is acceptable.

A COL applicant completing its detailed design and equipment selection during the COL design phase, must submit its specific D-RAP organization for staff review. This is COL Action Item 17.3.5-1.

17.3.6 SSC Identification/Prioritization

In CESSAR-DC Section 17.3.6, ABB-CE states that the PRA prepared for the System 80+ will be the primary source for identifying risk-significant SSCs that should be given special consideration during detailed design and procurement phases and should be considered for inclusion in the O-RAP. The method by which the PRA is used to identify risk-significant SSCs is described in CESSAR-DC Section 17.3.6. Table 17.3-4 of the CESSAR-DC gives the sections in CESSAR-DC where systems and equipment are specified to be included in the D-RAP. The primary source for the identification of systems and equipment to be included in the D-RAP is the PRA (Chapter 19 of the CESSAR-DC). It is also possible that some risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure data bases.

Section 17.3.6 of the CESSAR-DC contains a description of the analytical measures used to identify System 80+ risk-significant SSCs; risk achievement worth (RAW), risk reduction worth (RRW) and fussel-vesely worth (FVW). The primary analytical measure is the RAW which represents how the core damage frequency (CDF) would increase if an SSC always failed. RRW is a measure of how the CDF would be reduced if an SSC had a perfect reliability (i.e., never failed). The FVW is a measure of what fraction of the CDF the SSC failure contributes. CESSAR-DC Table 17.3-4 contains the locations for the descriptions, insights, and recommendations that resulted from the application of the risk-significant analytical measures and deterministic evaluations for internal events, external events, shutdown analysis, and engineering evaluations.

The staff reviewed CESSAR-DC Section 17.3.6 with respect to identifying and prioritizing risk-significant SSCs for the D-RAP. The staff concludes that CESSAR-DC Section 17.3.6 was responsive to the portions of RAI Q1, RAI Q2, DSER Open Item 17.3.1.2-1 associated with identifying and prioritizing risk- significant SSCs for the D-RAP meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk-significant SSCs as discussed in Section 17.3 of this report, and is acceptable. Therefore, CESSAR-DC 17.3.6 is acceptable. The staff's review and evaluation of the PRA methods or techniques to prioritize SSCs are addressed in Chapter 19 of this report.

17.3.7 Design Considerations

In CESSAR-DC Section 17.3.7, ABB-CE states that the reliability of risk-significant SSCs, which are identified by the PRA, will be evaluated at the detailed design stage by appropriate design reviews and reliability analyses. Current data bases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally, the failure rates will be similar, but in some cases they may differ because of recent design or data base changes. Whenever failure rates of designed risk-significant equipment are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve the appropriate reliability.

For those risk-significant SSCs, as identified by the PRA and other sources, component redesign (including selection of a different component) will be considered as a way to reduce the contribution to the CDF. If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following any redesign, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life (see Chapter 19 of this report).

Using the PRA or other design documents, ABB-CE will identify to the COL applicant the risk-significant SSCs, their associated failure modes and consequences, and reliability and availability assumptions, including any pertinent bases and uncertainties considered in the PRA (see Chapter 19 of
this report). ABB-CE will also provide this information for the COL applicant to consider in developing an O-RAP to assure that such SSCs can be expected to function with reliability commensurate with that assumed in the PRA. The COL applicant can use this information for establishing appropriate reliability and availability targets and the associated maintenance practices for achieving them.

ABB-CE has also stated that a COL applicant shall develop, as part of the D-RAP and O-RAP, a life-cycle management plan to aid in the design and operations activities intended to achieve the design life objectives. The life-cycle management plan is to be initiated early enough in the design completion process to (1) aid in the application, selection, and procurement of components with optimum design life characteristics, and (2) develop an aging management plan capable of assuring the plant's original design basis throughout its life.

ABB-CE also proposes that the aging management plan include specific components and mitigation measures. The aging management plan should be initiated early in the design process so that adequate provisions for mitigation measures can be made. In developing the life-cycle management plan, the COL applicant shall consider the design life requirements prescribed in Section 11.3, "Design Life," of the EPRI Utility Requirements Document and the insights gained from the Nuclear Plant Aging Research Program (e.g., NUREG/CR-4731 and NUREG/CR-5314).

The staff reviewed CESSAR-DC Section 17.3.7 with respect to using design reviews and reliability analyses during the detailed design stage and during plant operation. The staff concludes that CESSAR-DC Section 17.3.7 was responsive to the portion of RAI Q1 and DSER Open Item 17.3.1.2-1 associated with providing a process for evaluating risk-significant SSCs for redesign and a process for providing information to a COL applicant for establishing appropriate reliability targets and the associated maintenance practices for an O-RAP. The staff concludes that the life-cycle management plan proposed by ABB-CE is consistent with the goals and objectives of RAP. The staff also concludes that CESSAR-DC Section 17.3.7 meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk-significant SSCs as discussed in Section 17.3 of this report, and is acceptable.

17.3.8 Defining Failure Modes

In CESSAR-DC Section 17.3.8, ABB-CE states that the determination of dominate failure modes of risk-significant SSCs will include historical information, analytical models, and existing requirements. Many PWR systems and components have compiled a significant historical record. For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary.

ABB-CE uses the methodology of NUREG/CR-5695, "A Process for Risk-Focused Maintenance" Section 5, to determine dominant failure modes of risk-significant SSCs in the D-RAP. The staff reviewed CESSAR-DC Section 17.3.8 with respect to the methodology used to determine dominant failure modes of risk-significant SSCs in the D-RAP. The staff concludes that CESSAR-DC Section 17.3.8 was responsive to the portion of RAI Q1 and DSER Open Item 17.3.1.2-1, associated with the methodology used to determine dominant failure modes of risk-significant SSCs in the D-RAP. This information satisfies the requirement of the applicable regulation for D-RAP to define failure modes as described in Section 17.3 of this report, and is acceptable.

17.3.9 Operational Reliability Assurance Activities

In CESSAR-DC Section 17.3.9, ABB-CE states that once the dominant failure modes are determined for risk-significant SSCs, an assessment should be used to determine suggested O-RAP activities that will ensure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, or periodic preventive maintenance. Some SSCs may require a combination of activities to ensure that their performance is consistent with the PRA.

Periodic testing of SSCs may include startup of standby systems; surveillance testing of instrument circuits to ensure that they will respond to appropriate signals; and inspection of passive SSCs to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurements of output, measurement of magnitude of an important variable, and testing for abnormal conditions. Periodic preventive maintenance (PM) is an activity performed at regular intervals to preclude problems that could occur before the next PM interval.

Planned maintenance activities should be integrated with regular operating plans so that they do not disrupt normal operation. Maintenance that will be performed more frequently than refueling outages must be planned to avoid disrupting safe operation or causing a reactor scram, engineered safety feature actuation, or abnormal transient. Maintenance planned for performance during refueling outages must be conducted in such a way that it will not adversely affect plant safety.

The staff reviewed CESSAR-DC Section 17.3.9 with respect to the process to determine operational reliability assurance activities using the dominant failure modes identified in the D-RAP. The staff concludes that CESSAR-DC Section 17.3.9 was responsive to the portion of RAI Q2 associated with the process to determine operational reliability assurance activities using the dominant failure modes identified in the D-RAP and is acceptable. The COL applicant should also incorporate the O-RAP objectives into existing programs such as maintenance and quality assurance and provide the staff with a description of how these objectives are met at the time of the COL application. This is COL Action Item 17.3.9-1.

17.3.10 Combined License Applicant's Reliability Assurance Process

In CESSAR-DC Section 17.3.10, ABB-CE states that the O-RAP that will be prepared and implemented by the COL applicant should make use of information submitted by ABB-CE. The information will help the COL applicant determine activities that should be included in the O-RAP. Examples of elements that might be included in an O-RAP are as follows:

- reliability performance monitoring
- reliability methodology
- problem prioritization
- root cause analysis
- corrective action determination
- corrective action implementation
- corrective action verification
- plant aging
- feedback to designer
- programmatic interfaces
- maintenance rule (10 CFR 50.65) integration

These elements are also defined in CESSAR-DC Section 17.3.10. The COL applicant will address in its O-RAP the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, QA, and procurement of replacement equipment.

The staff reviewed CESSAR-DC Section 17.3.10 with respect to the COL applicant's reliability assurance program elements and definitions of these elements. The staff concludes that CESSAR-DC Section 17.3.10 is responsive to the portion of RAI Q2, RAI Q3, RAI Q5, RAI Q6, and DSER Open Item 17.3.3.3-1 associated with the COL applicant's reliability assurance program elements and program element definitions and is acceptable. The COL applicant will submit the D-RAP for completion of the detailed design and specific equipment selection phases (e.g., procurement of risk-significant SSCs). The COL applicant should also incorporate the O-RAP objectives into existing programs such as maintenance and quality assurance and provide the staff with a description of how these objectives are met. These are included in COL Action Items 17.3.1-1 and 17.3.9-1 above.

17.3.11 D-RAP Implementation

In CESSAR Section 17.3.11, ABB-CE provided an example of implementation of the D-RAP for the CCWS. This system was selected as an example because it is not a front-line safety system but was found in the earlier System 80 PRA to contain risk-significant components. Using that finding, the D-RAP organization described in CESSAR-DC Section 17.3.5, changed the system design for System 80+. The design change and analytical process are described in this chapter, are presented as a D-RAP example only, and do not necessarily correspond to the current System 80+ design.

The staff concludes that the CESSAR-DC Section 17.3.11 example using the CCWS satisfactorily demonstrated ABB-CE's understanding of the RAP concept and their ability to implement it into the design. This information is responsive to the staff's RAI question, and is acceptable. The process description for the implementation of the System 80+ D-RAP by a COL applicant will be reviewed by the staff at the time a COL application is submitted.

17.3.12 Conclusion

The staff has reviewed Section 17.3 of the CESSAR-DC. The staff finds that the CESSAR-DC satisfies the requirements of the applicable regulation for the reliability assurance program for the design phase of the System 80+, as stated above, and is acceptable.

18 HUMAN FACTORS ENGINEERING

Executive Summary

This chapter describes the staff's final evaluation of the human factors engineering (HFE) specifications presented in Chapter 18 of the ASEA Brown Boveri-Combustion Engineering (ABB-CE) Standard Safety Analysis Report (CESSAR-DC). This section also addresses aspects of the training and plant procedures presented in CESSAR-DC Sections 13.2 and 13.5, as well as additional materials that ABB-CE submitted related to human factors engineering (HFE).

The staff based its review on current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f); Standard Review Plan (SRP) Sections 13 and 18; NUREG-0700, "Guidelines for Control Room Design Reviews," and NUREG-0933, "A Prioritization of Generic Safety Issues." The staff also developed review criteria for aspects of the System 80+ HFE program that were not fully addressed by these documents. These criteria represent a slightly modified version of the "HFE Program Review Model and Acceptance Criteria" (HFE PRM), which the staff forwarded to the Commission in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

The staff conducted its review in two phases. The staff's preliminary review of early versions of the CESSAR-DC was documented in NUREG-1462, the draft safety evaluation report (DSER), dated September 1992. The staff then reviewed DSER issue resolution and further development of the CESSAR-DC. As part of the final review, the staff modified the HFE PRM identified above for the evaluation of a design process. The review focused on (1) resolution of DSER issues, and (2) evaluation of the ABB-CE documents with respect to the topics and general criteria of the HFE PRM. The following major topics were addressed:

- human factors engineering program management
- operating experience review
- functional requirements analysis and allocation
- task analysis
- human/system interface (HSI) design
- plant and emergency operations procedures
- verification & validation (V&V)
- certified design description/inspections, tests, analyses, and acceptance criteria (CDD/ITAAC)

The findings in each area are summarized below.

Human Factors Engineering Program Management

HFE PRM Element 1, "Human Factors Engineering Program Management," specified that a formal HFE program should be established to guide HFE activities. The staff's DSER review of the CESSAR-DC identified several issues related to HFE PRM Element 1. These were designated as DSER Issues 18.3.1-1, 18.3.2-1, and 18.3.5-1.

The ABB-CE human factors engineering program plan (HFPP) and related sections of the CESSAR-DC developed in response to the staff's DSER issues acceptably address the criteria in HFE PRM Element 1, and the DSER issues are resolved. Although the HFPP does not include procedure development as part of its technical program, ABB-CE modified the CESSAR-DC to incorporate a COL action item to address aspects of procedure development that were required by the HFE PRM. This the COL action item is acceptable, and the issue is resolved.

Operating Experience Review

HFE PRM Element 2, "Operating Experience Review," specified that operating experience should be reviewed. The staff's DSER review of the CESSAR-DC identified operating experience (OER) review as DSER issue 18.4.

The staff evaluated the original "ABB-CE Operating Experience Review for System 80+ MMI Design" using the HFE PRM criteria. Overall, the ABB-CE OER was quite impressive. It showed a detailed review of many aspects of pertinent commercial nuclear power plant experience, and incorporated appropriate design features into the System 80+ design. Not all aspects of the HFE PRM were completely addressed. ABB-CE worked with the staff to address the designated concerns related to areas of operating experience and added items to the HF issues tracking system for later consideration of incorporation into the design. This additional work is discussed in a revised OER, which also better describes how ABB-CE had already incorporated operating experience into the System 80+ design. The revised ABB-CE OER meets the criteria in the HFE PRM.

Functional Requirements Analysis and Allocation

HFE PRM Element 3, "Functional Requirements Analysis," and Element 4, "Function Allocation," specified that an analysis of functional requirements and a structured and documented allocation of functions should be conducted. ABB-CE has stated that full analyses of functional requirements and function allocation are not necessary because the

System 80+ design is an evolution of the System 80 design that was previously reviewed and approved by the NRC and has an operating history (at Palo Verde Units 1, 2, and 3). In addition, ABB-CE has stated that the definition and allocation of functions for the System 80+ design are largely unchanged from that of its predecessor, the System 80 design. The staff agreed, and the HFE PRM criteria were modified accordingly.

ABB-CE's "Human Factors Evaluation and Allocation of System 80+ Functions" describes the critical functions and the success paths that are responsible for satisfying the safety functions. This document makes high-level comparisons and notes differences between the System 80 and the System 80+ designs. The ABB-CE document is a useful information source that describes the basic structure of the System 80+ plant, the operator's role that results from this basic structure, and the similarities and differences between the System 80+ and System 80 designs with respect to these basic structures and operator's role.

Differences in success paths were noted and the following critical safety function (CSF) success paths were identified as added:

- rapid depressurization (reactor coolant system (RCS) heat removal function)
- hydrogen ignitors (containment environment function)

the following CSF success paths were identified as modified:

- alternate generator (vital auxiliaries function)
- startup feed (RCS heat removal)

No CSF success paths were identified as deleted. ABB-CE has stated that the identified additions and modifications should impose little difference in the operator's role in the System 80+ design compared to the System 80 design.

The function allocation analyses provided by ABB-CE in "Human Factors Evaluation and Allocation of System 80+ Functions" were done after much of the function allocation between personnel and plant systems had been completed for the System 80+ design. The ABB-CE report thus provided documentation and justification of the function allocation that had emerged from the System 80+ design process, rather than an analysis of an allocation process that was "in progress." The justifications for function allocations were found to be acceptable. Some specific information requirements of the HFE PRM were not provided in "Human Factors Evaluation and Allocation of System 80+ Functions," but were adequately addressed by ABB-CE's task analysis methodology.

In summary, the staff finds ABB-CE's functional requirements analysis and allocation acceptable based upon the HFE PRM criteria.

Task Analysis

HFE PRM Element 5, "Task Analysis," specified that a task analysis should be conducted. ABB-CE described its task analysis methodology in Section 18.5 of the CESSAR-DC and in "System 80+ Function and Task Analysis Final Report" (dated January 1989, docketed April 8, 1992). This methodology is referred to in this section as functional task analysis (FTA). In the DSER the staff identified deficiencies in the scope and depth of analyses provided by ABB-CE. This was identified as DSER Issue 18.7. In response to this DSER issue, ABB-CE submitted its proposed revision for CESSAR-DC Section 18.5 (Amendment Q).

The revised methodology, referred to as standard safety analysis report functional task analysis (SSARFTA), adequately addresses control and display requirements. The proposed scope of the SSARFTA effort addresses an adequate range of system failures and plant operating conditions and includes critical tasks specified in probabilistic risk assessment (PRA). Acceptable categories of control and display requirements are generated for individual HSI components, including device type, range, accuracy, precision, and units of measure. In addition, the task analysis includes provisions for recording special support characteristics necessary for facilitating operator tasks.

The revised task analysis methodology adequately evaluates performance requirements imposed on operators, and assesses task loading by determining whether the time required for task element completion is consistent with the time available for completion. The criteria for estimating the human performance time were revised by ABB-CE as a result of this review, and resulting criteria are acceptable. Although this methodology adequately evaluated operator performance at the individual task element level, some DSER issues and HFE PRM criteria regarding communication and coordination between multiple crew members are not evaluated. However, these concerns are adequately addressed by ABB-CE's verification and validation program. As a result, the staff finds the task analysis methodology acceptable.

Human/System Interface Design

HFE PRM Element 6, "Human-System Interface Design," specifies that HFE principles and criteria should be applied along with other design requirements to identify, select, and design the particular equipment to be operated/, maintained, and controlled by plant personnel. Element 6 concerns design methods, criteria used for making design decisions, interim products such as standard design features, and the final design. (HFE PRM Element 8, "Verification and Validation," provides a detailed review of the final design.)

Issues related to Element 6 were addressed in three major reviews, pertaining to standard design features; design methods and general characteristics; and human factors engineering standards, guidelines, and bases (HFESGB). Each of these distinct phases of the HFE PRM Element 6 review is briefly described below.

Standard Design Features

The standard design features review evaluated important elements of the Nuplex 80+ design, including six standard design features and the integrated process status overview (IPSO). This review focused on the acceptability of these features as standard design elements, as described in the CESSAR-DC and other design basis documents, and as represented in the mockups of the master control console (MCC) and IPSO. The objective of the review was to determine the acceptability of the basic design features of the System 80+ advanced control room, as described in the CESSAR-DC and other design basis documents, with regard to their consistency with established HF standards, guidelines, and principles. Further, the control room design was reviewed against Supplement 1 to NUREG-0737 requirements for a safety parameter display system (SPDS).

The staff found that the seven design features addressed by this review are consistent with HFE design principles and guidelines, and that the HSI design adequately addressed SPDS criteria. In some cases, specific issues could not be resolved at this stage of the HSI design. ABB-CE recorded these in the HF issue tracking system, and has committed to resolve these issues in later stages of the design and evaluation process.

ABB-CE's design justifications throughout the design features review adhere to HF guidelines (as presented in ABB-CE's HFESGB document) and subjective evaluations. The review revealed a high degree of consistency between the design and these guidelines. However, some issues regarding human performance are not directly addressed by available guidelines. ABB-CE's efforts to evaluate human performance issues during the development of the HSI via System 80+ specific experiments or other analyses that measure human performance have been very limited. ABB-CE has committed to evaluate issues of human performance related to its final, integrated HSI design during its verification and validation effort. These evaluations will use mockups and simulators. This effort is addressed by HFE PRM Element 8, "Verification and Validation." In addition, ABB-CE has committed to evaluate issues related to the alarm system using a prototype of the discrete indication and alarm system (DIAS) before verification and validation. Based on the staff's review of ABB-CE's design features and ABB-CE's commitments to evaluate human performance issues in later stages of the design process, the design features are acceptable.

HSI Design Methods and General Characteristics

The design methods and general characteristics review evaluated

- the methods for implementing the display and control requirements, selecting hardware and software, and refining design concepts
- criteria used to determine control room and control panel arrangements including the overall configuration of the main control console and the position of individual control/display devices within individual panels
- general design characteristics that were incorporated into the HSI

The application of the methods and criteria to the design of the control room configuration, RCS panel, and remote shutdown panel as well as, relevant DSER issues were evaluated in this review. These considerations were evaluated within the context of the main control room configuration, the presentation of information on controls and displays, and the layout of panels. Specific attention was given to the RCS panel and the remote shutdown panel.

The staff found the application of methods, design criteria, and general design characteristics acceptable. However, the staff identified specific issues related to concerns information presentation, panel layout, and configuration. ABB-CE provided responses and commitments via its HF issue tracking system to address these issues in later stages of the design process. The most significant of ABB-CE's commitments were to provide more detailed descriptions of the human/system interface to support the following:

- data entry tasks
- blocking and tagging tasks via the data processing system (DPS) and the DIAS of instrumented and non-instrumented components

- operator established alarms
- component control system (CCS) operator module

ABB-CE made additional commitments as a result of the staff's review.

HFE Standards, Guidelines, and Bases (HFESGB)

The HFESGB review evaluates ABB-CE's HFE design criteria used to identify, select, and design equipment to be operated, maintained, and controlled by plant personnel with respect to accepted HF guidance and practices. This review primarily addressed ABB-CE's "Human Factors Engineering Standards, Guidelines, and Bases for Nuplex 80+," and the technical basis and validity, level of detail, integration, and procedure for implementation of the stated design guidelines.

This review concluded that the HFESGB has an acceptable scope that includes aspects of the HSI, both inside and outside the main control room, that are important to safe operation and maintenance of the plant by personnel. The review also found that the HFESGB includes general design guidance that was derived from acceptable HF source documents. The guidance provided by the HFESGB was presented at a level of detail that was appropriate for many of the design areas addressed. However, in some cases, specific guidance was lacking with respect to unique areas of the System 80+ HSI design. ABB-CE committed to include additional guidance in the HFESGB to address these areas.

The review identified a lack of procedures or other guidance for the systematic implementation of the HFESGB guidelines and standards for the design of DPS displays. Although procedures for other design activities such as control room layout and panel layout are well defined in CESSAR-DC, neither the HFESGB nor the CESSAR-DC provides similar procedures for the application of HFESGB guidance to the design of DPS displays. ABB-CE addressed this concern by committing in its HFPP to use a systematic process for display design. The application of this systematic process can be verified through documentation showing the results of design reviews, application of FTA results, and checklists of important characteristics for each display page.

The review revealed some differences between specific design criteria provided by the HFESGB and design criteria recommended by accepted HF literature. These were addressed and resolved on a case-by-case basis.

ABB-CE's commitments, in response to this review, adequately resolve the general concerns of scope, technical basis and validity, level of detail, and procedure for implementation. On that basis, the HFESGB was found to be a generally acceptable source of HF guidance for the design of the System 80+ HSI.

Plant and Emergency Operations Procedures

HFE PRM Element 7, "Plant and Emergency Operations Procedure Development," specified that procedures should be developed as part of the HFE effort. The objective of this review was to ensure that HFE principles and criteria are applied along with all other design requirements to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. The staff's DSER review of the CESSAR-DC identified DSER Issues 18.9.1, "Operating Support Information Program," (OSIP), 18.9.2, "Emergency Procedure Guidelines," and 20.2-3 (Issue I.C.1, ABB-CE committed to modify emergency operations guidelines (EOGs) ensuring compatibility with System 80+ design).

Detailed procedure development and validation is identified in the CESSAR-DC as a COL action item, which will use information that will be provided by ABB-CE as part of the HFE program. The staff found that ABB-CE's approach to System 80+ procedure development is acceptable.

Verification and Validation (V&V)

HFE PRM Element 8, "Verification and Validation," specifies that a formal V&V of the HSI should be performed. The staff's DSER review of the CESSAR-DC identified DSER Issue 18.10-1, related to HFE PRM Element 8.

The ABB-CE approach to V&V has been reviewed and found acceptable. Although the present V&V plan lacks complete methodological detail, a more detailed implementation plan will be developed following design certification. Requirements for the additional detail, addressing staff concerns, are provided in Appendix B of the plan. This approach is acceptable to the staff, because V&V details are more appropriately addressed in a detailed implementation plan, which can best be developed when the design becomes complete.

Certified Design Description/Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

The objective of this review was to evaluate the System 80+ main control room ITAAC, remote shutdown room ITAAC, and control panel ITAAC against the requirements of 10 CFR Part 52.47(a)(1)(vi).

The staff concluded that the System 80+ HFE design and implementation processes as described in the CDD and CESSAR-DC are acceptable. The Tier 2 commitments described in the System 80+ CESSAR-DC and related (docketed) documents provide methods and descriptions of the implementation of the Tier 1 requirements. The determination that the plant has been constructed in accordance with the design certification will require use of the information contained in both the Tier 1 and Tier 2 documents. The Tier 2 material contained in the following System 80+ CESSAR-DC sections was used to support the safety finding with regard to the design and implementation process:

- Section 18.5, "Functional Task Analysis"
- Section 18.6, "Control Room Configuration"
- Section 18.7, "Information Presentation and Panel Layout Evaluation"
- Section 18.8, "Control and Monitoring Outside the Main Control Room"
- Section 18.9, "Verification and Validation"

Per SECY-92-287, any change to commitments by the COL applicant regarding the above CESSAR-DC sections would involve an unreviewed safety question and, therefore, would require NRC review and approval before implementation. Any requested change to the subject CESSAR-DC section commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

Conclusions

The staff reviewed the HFE process described by ABB-CE in the CESSAR-DC and CESSAR-DC-referenced documents. Based on its review, the staff concludes that the ABB-CE HFE program is acceptable and will result in acceptable HSI designs for the main control room, remote shutdown system, and related applicable HSIs. The basic design features of the System 80+ advanced control room were reviewed and found consistent with HF standards, guidelines, and principles, and are acceptable for use in the control room. In addition, the staff concludes that the design commitments and HFE ITAAC and DAC accurately summarize the minimum HFE requirements for an acceptable design and verification/validation of the main control room and remote shutdown room. All previously identified DSER issues have been adequately addressed and are resolved.

Introduction

This chapter describes the staff's final evaluation the human factors engineering (HFE) specifications presented in Chapter 18 of the ABB-CE CESSAR-DC. This section also addresses aspects of the training and plant procedures presented in CESSAR-DC Sections 13.2 and 13.5, as well as additional materials that ABB-CE submitted related to human factors engineering (HFE). All materials used in this evaluation are described in the subsections of the technical review. The staff based its review on current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f); SRP Sections 13 and 18; NUREG-0700, "Guidelines for Control Room Design Reviews"; and NUREG-0933, "A Prioritization of Generic Safety Issues." The staff also developed additional review criteria for aspects of the System 80+ HFE program that were not fully addressed by these documents. These criteria represent a slightly modified version of the HFE PRM, which the staff forwarded to the Commission in SECY-92-299, dated August 27, 1992. Section 18.1.3 presents an overview of the HFE PRM.

The System 80+ standard design includes the Nuplex 80+ advanced control complex, which is divided into several functional units including the main control room. Nuplex 80+ consists of numerous interdependent systems such as the main control panels and remote shutdown panel. CESSAR-DC Section 1.2.6. provides a detailed discussion of the relationship between Nuplex 80+ and System 80+.

Section 18.1 of this report describes the methodology used to conduct the evaluation, including the development of general review criteria that supplement the regulatory requirements and established guidelines. The results described in Sections 18.2 through 18.10 address the following major topics, respectively:

- human factors engineering program management
- operating experience review
- functional requirements analysis and allocation
- task analysis
- human/system interface design
- plant and emergency operating procedures
- verification and validation
- certified design description/inspections, tests, analyses, and acceptance criteria

Section 18.11 summarizes the evaluation findings and overall conclusions.

During its initial review of the CESSAR-DC, the staff identified and documented many outstanding issues in the DSER. One of the major issues to emerge from the initial review was that detailed HSI information concerning the final design was not available for staff review as part of the design certification evaluation. ABB-CE's HSI analysis and design efforts provided a list of standard design features characterized at a general level (not a detailed specification) and a minimum inventory of fixed safety-significant information and control requirements derived from an analysis of the EOGs and PRA. Evaluation of the standard features and the inventory are part of the certification review. However, development of standard features is part of an ongoing design process that has not reached the stage of detailed implementations for the control room panels. By themselves, the descriptions of the standard design features do not provide a basis upon which a safety determination can be made.

In SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," the staff proposed the use of design acceptance criteria (DAC) as an approach to the review portions of the System 80+ design when detailed design information was unavailable due to rapidly changing technology. This included HF aspects of the control room and remote shutdown station design.

DAC are prescribed limits, parameters, procedures, and attributes upon which the NRC relied to make a final safety determination to support design certification. The DAC are measurable or testable and must be verified in order for the staff to accept the final design. DAC delineate the process and requirements that a combined license (COL) applicant must implement during the development of detailed design information for the control room and the remote shutdown station. A number of conformance review points are specified to periodically assess the adequacy of the detailed design as it develops.

Because the criteria for review of a design and implementation process were not clearly defined in current regulations and guidance documents, the staff developed criteria as part of this review. These criteria provided the basis to (1) assess whether the appropriate HFE elements are included in the design and implementation process, (2) identify what materials need to be reviewed for each element, (3) evaluate the proposed DAC and ITAAC to be used by the staff to verify each of the review elements, and (4) assess the adequacy of the DAC and ITAAC developed by ABB-CE.

The staff's design certification evaluation is based partially on design information and partially on an implementation process plan that describes the HFE program elements required to develop the key features and inventory into an acceptable detailed design specification. Along with the design and implementation process, ABB-CE has provided the necessary DAC and ITAAC to ensure that the design and implementation processes are properly executed by the COL applicant. ABB-CE has submitted a design and implementation process for the major design activities for the System 80+ HFE effort. The staff specified that the design and implementation process will contain descriptions of all HF activities (elements) that are necessary and sufficient for the development and implementation of the System 80+ HSI that will protect the health and safety of the public.

18.1 General Methodology

The staff conducted its review in two phases. The staff's preliminary review of early versions of the CESSAR-DC was documented in NUREG-1462, the DSER, dated September 1992. Section 18.1.1 summarized this review. The staff then reviewed DSER issue resolution and further development of the CESSAR-DC. Section 18.1.2 describes the scope of this subsequent review. As part of the final review, the staff modified the HFE PRM for the evaluation of a design process. Section 18.1.3 describes the model development. The following sections present the detailed review criteria as they relate to the preliminary review and DSER issues, as well as, the final SSAR review, and describe the objectives and rationale for development of those criteria.

18.1.1 Preliminary Review and Draft Safety Evaluation Report Issues

The primary sources of information reviewed by the staff for the DSER were CESSAR-DC Chapter 18 and ABB-CE's responses to staff requests for additional information (RAI), designated as Questions (Q)620.1 through Q620.38.

The review focused on the aspects of ABB-CE's HF considerations outlined above. In addition, the review included ABB-CE's resolution of various safety issues (unresolved safety issues, generic safety issues, and the construction permit/manufacturing license rule of 10 CFR 50.34(f)) related to HF considerations addressed in CESSAR-DC Chapters 13 and 18.

From its initial review, the staff concluded that the HF program for the HSI did not provide sufficient information to support a determination that the System 80+ design as proposed by ABB-CE for certification would adequately incorporate accepted HF considerations in a manner that would achieve required safety and reliability. The staff cited the following principal reasons for this finding were:

- Design bases were specified without supporting rationale.

- A design process was presented in insufficient detail and without results. The HSI design requirements were presented without evidence that they were derived from the design process and without supporting tests/evaluations.

- The documentation did not provide sufficient detail to support the review of the System 80+ HF efforts to a level necessary for design certification.

Specific issues identified as requiring resolution are identified in Table 18.1, which also indicates the FSER section which addresses the DSER issue.

18.1.2 Final Standard Safety Analysis Report Review

The primary sources of information used for the final review described in this chapter were CESSAR-DC Chapter 18 and ABB-CE's responses to the DSER issues. As the DSER issues were resolved, ABB-CE provided much additional documentation addressing staff concerns. Much of this information was provided in the form of docketed technical plans and analysis reports. (A complete list of the materials relied on for preparation of the FSER appears in each of the review sections presented below.) The staff's review of these materials, as well as revisions to the CESSAR-DC, gave rise to additional questions which were resolved through numerous public meetings and documented telephone conversations between the staff and ABB-CE. The issues raised and their resolutions are described in detail in the sections below.

In addition to the evaluation of ABB-CE documentation, the design certification review was supported by information obtained from on-site reviews conducted using mock-ups of the System 80+ control room design and interviews with operators of System 80 plants.

18.1.3 Development of Review Criteria

18.1.3.1 Objectives

Since all details of the final design were not available for review, certification is based partially on the staff's approval of a design and implementation process plan. In order for ABB-CE's design and implementation process to result in an acceptable design, it must contain (1) descriptions of all required HFE program elements for the design, development, and implementation of the System 80+ HSI, and (2) DAC for the reviews under ITAAC.

To review ABB-CE's proposed HFE process, the staff was required to (1) assess whether all of the appropriate HFE elements were included, (2) identify what materials needed to be reviewed for each element, and (3) evaluate the proposed DAC and ITAAC to verify each of the elements. To conduct the review, the staff identified which aspects of the HSI design process were required to ensure that HFE safety goals are achieved, and identified the review criteria by which each element can be assessed. Review criteria independent of those provided by ABB-CE were required to ensure that ABB-CE's plan reflects currently accepted HFE practices and is thorough, complete, and workable. To support such a review, the staff developed a technical basis for review of the HSI design process. The following specific objectives guided this effort:

(1) Develop an HFE PRM to serve as a technical basis for the review of the process proposed by ABB-CE for certification. The model needed to be (a) based on currently accepted HFE practices, (b) well-defined, and (c) validated through experience with the development of complex, high-reliability systems.

(2) Identify the HFE elements in a system development, design, and evaluation process that are necessary and sufficient requisites to successful integration of the human component in complex systems.

(3) Identify which aspects of each HFE element are key to a safety review and are required in order to monitor the process implementation.

(4) Specify the acceptance criteria by which HFE elements can be evaluated as design development progresses.

18.1.3.2 HFE PRM Development

The staff reviewed current HFE guidance and practices described in a wide range of nuclear and non-nuclear industry documents to identify important HFPP elements relevant to a design process review. A generic systems development, design, and evaluation process was defined with eight key HFE elements that included criteria by which they could be assessed. This is referred to as the HFE PRM, which was based largely on applied general systems theory and the Department of Defense (DOD) systems development process (which is rooted in systems theory). Applied general systems theory provides a broad approach to system design and development, based on a series of clearly defined developmental steps, each with clearly defined goals and specific management processes to attain them. Systems engineering has been defined as "the management function which controls the total system development effort for the purpose of achieving an optimum balance of all system elements. It is a process which transforms an operational need into a description of system parameters and integrates those parameters to optimize the overall system effectiveness" (Kockler, F. et al., 1990).

Use of the DOD systems development process and procedure as an input to the development of the HFE PRM was based on several factors. DOD policy identifies personnel

Table 18.1 Chapter 18 DSER issues

Issue No.	Issue	DSER Section	FSER Section Where Addressed
13.1-1	The COL applicant referencing the System 80+ Standard Plant Design will be required to provide site-specific information at the COL phase described in 10 CFR 52.79(b)	13.1	13.1
18.3.1	Human Factors Engineering Program Plan	18.3.1	18.2.3.1
18.3.2	HFE Program Milestones and Task Schedules	18.3.2	18.2.3.1
18.3.5	Design Goals	18.3.5	18.2.3.1
18.4	Operating Experience Review	18.4	18.3.3.1
18.5.1	Identification and Traceability of Human Factors Requirements	18.5.1	18.2.3.1
18.5.2	Function Analysis	18.5.2	18.4.4.2
18.6	Function Allocation	18.6	18.4.5.2
18.7	Task Analysis	18.7	18.5.3
18.8	Human-System Interface Design	18.8	18.6.3.3.1
18.8.1	Shape Coding Used to Prioritize Alarms	18.8.1.1	18.6.1.3.3
18.8.1.3	Flash Coding of Alarms	18.8.1.3	18.6.1.3.3
18.8.1.4	Size Coding of Alarms	18.8.1.4	18.6.1.3.3
18.8.1.5	Quantity and Types of Information Encoded in the Control Room	18.8.1.5	18.6.3.3.1
18.8.2	Additional Information Required for Staff Review	18.8.2	18.6
18.9.1	Operational Support Information Program	18.9.1	18.7.3.1
18.9.2	Emergency Operations Guidelines	18.9.2	18.7.3.1
18.10	Human Factors Verification and Validation	18.10	18.8.3.1

Issue No.	Issue	DSER Section	FSER Section Where Addressed
20.2-3	Issue I.C.1: ABB-CE committed to modify EPGs ensuring compatibility with System 80+ design	20.2	18.7.3.1
20.2-4	Issue 83 (Control Room Habitability)	20.2	20.3
20.2-10	Issue I.A.1.4 (Long-Term Upgrading of Operating Personnel and Staffing)	20.2	18.3.3.2.5 20.4
20.2-11	Issue I.C.9 (Long-Term Program for Upgrading Procedures)	20.2	18.3.3.2.5 13.5 20.4
20.3-1	TMI Action Item I.A.4.2 (Simulator Capability) and Item II.J.3.1 (Management Plan for Design and Construction Activities)	20.3	18.3.3.2.5 20.4
20.2-17	Issue 125.I.3 (Safety Parameter Display System Availability)	20.2	18.3.3.2.5 18.6.1.3.4 20.3
20.1-19	Issue B-17 (Criteria for Safety-Related Operator Actions)	20.1	18.3.3.2.5 20.2
20.2-21	Issue I.C.1 (Guidance for Evaluation and Development of Procedures for Transients and Accidents)	20.2	18.3.3.2.5 13.5 20.4
20.2-22	Issue I.D.2 (Plant Safety Parameter Display System Console)	20.2	18.3.3.2.5 20.4
20.2-23	Issue I.D.4 (Control Room Design Standard)	20.2	18.3.3.2 20.4
20.2-24	Issue I.D.5(1) (Control Room Design - Improved Instrumentation Research Alarms and Displays)	20.2	18.3.3.2.3 20.4
20.2-27	Issue II.K.1.5 (Safety-Related Valve Position Description)	20.2	18.3.3.2.5 20.4
20.2-28	Issues I.D.3 and II.K.1.10 (Review and Modify Procedures for Removing Safety-Related Systems from Service)	20.2	18.3.3.2.5 20.4
20.2-29	Issue HF1.3.4c: MMI-Operational Aids	20.2	18.3.3.2.5 20.5
	Issue HF1.3.4d: MMI-Automation & Artificial Intelligence	20.2	18.3.3.2.5 20.5
	Issue HF1.3.4e: MMI-Computers & Computer Displays	20.2	18.3.3.2.5 20.5

Issue No.	Issue	DSER Section	FSER Section Where Addressed
	Issue HF1.4.4: Guidelines for Upgrading Other Procedures	20.2	18.3.3.2.5 13.5 20.5
	Issue HF5.1: Local Control Station	20.2	18.3.3.2.5 20.5
	Issue HF5.2: Review Criteria for Human Factors Aspects of Advanced Controls & Instrumentation	20.2	18.3.3.2.5 20.5
	Issue HF1.1: Shift Staffing	20.2	18.3.3.2.5 20.5
	Issue HF1.3.4a: MMI-Control Stations	20.2	18.3.3.2.5 20.5
	Issue HF1.3.4b: MMI-Annunciators	20.2	18.3.3.2.5 20.5

as a specific component of the total system. A systems approach implies that all system components (hardware, software, personnel, support, procedures, and training) are given adequate consideration in the developmental process. A basic assumption is that the personnel component receives serious consideration from the very beginning of the design process. In addition, the DOD, compared with non-military system developers, has the most experience in applying HFE to the development of complex, technical systems; thus, its process is evolved and formalized, and it represents the most highly developed and well-defined model of the HFE process available.

Within the DOD approach, the development of a complex system begins with the mission or purpose of the system and the capability requirements needed to satisfy mission objectives. Systems engineering is essential in the earliest planning period to develop the system concept and to define the system requirements. During the detailed design of the system, systems engineering ensures

- balanced influence of all required design specialties
- resolution of interface problems
- effective conduct of trade-off analyses
- effective conduct of design reviews
- verification and validation of system performance

The effective integration of HFE considerations into the design is accomplished by providing (1) a structured top-down system development approach that is iterative, integrative, and interdisciplinary, and (2) a management structure that details the HFE considerations in each step of the overall process. A structured top-down approach to nuclear power plant (NPP) HFE is consistent with the approach to new control room design, as described in Appendix B to NUREG-0700 and the more recent internationally accepted industry standard, International Electrotechnical Commission 964, "International Standard: Design for Control Rooms of Nuclear Power Plants," for advanced control room design. The approach is also consistent with the recognition that HF issues and problems emerge throughout the NPP design and evaluation process; therefore, HF issues are best addressed with a comprehensive top-down program.

The scope of the HFE PRM excluded a training program development element, because training is adequately addressed by existing NRC requirements. In addition, human reliability analysis was excluded and is addressed in Chapter 19 of this report.

NRC HFE requirements of 10 CFR 50.34(f)(2)(iii) were incorporated into the HFE PRM, as required by 10 CFR 52.47(a)(1)(ii).

18.1.3.3 Model Description

The model is intended as the programmatic approach to achieving a design commitment to HFE. The overall commitment and scope of the HFE effort can be stated as follows: HSIs shall be provided for the operation, maintenance, test, and inspection of the System 80+ that reflect "state-of-the-art human factors principles" (10 CFR 50.34(f)(2)(iii)) as required by 10 CFR 52.47(a)(1)(ii). For the purposes of the model development "state-of-the-art" HF principles were defined as those principles currently accepted by human factors practitioners. "Current" is defined with reference to the time at which this model was developed. "Accepted" is defined as a practice, method, or guide that is (1) documented in the HF literature within a standard or guidance document that has undergone a peer-review process and/or (2) justified through scientific/industry analysis, design, and evaluation practices.

All aspects of HSI will be developed, designed, and evaluated on the basis of a structured top-down system analysis using accepted HFE principles based on current HFE practices. HSI is used here in the very broad sense and shall include all operations, maintenance, testing, and inspection interfaces and procedures materials.

The model developed to achieve this commitment contains the following eight elements:

- Element 1 human factors engineering program management
- Element 2 operating experience review
- Element 3 system functional requirements analysis
- Element 4 allocation of function
- Element 5 task analysis
- Element 6 human/system interface design
- Element 7 plant and emergency operations procedure development
- Element 8 human factors verification and validation

Element 1 Human Factors Engineering Program Management

To ensure the integration of HFE into system development and the achievement of the goals of the HFE effort, an HFE design team and an HFE program plan shall be established to ensure the proper development, execution, oversight, and documentation of the HFE program. As part of the program plan, an HFE issue tracking system (to document and track HFE related problems, concerns, and issues and their solutions throughout the HFE program) will be established. The HFE issue tracking system was used in the evaluations (Sections 18.2 through 18.8 of this report) as a mechanism to log specific design issues.

Element 2 Operating Experience Review

The accident at Three Mile Island (TMI) in 1979 and other reactor incidents have illustrated significant problems in the actual design and design philosophy of NPP HSIs. Many studies have been conducted as a result of these incidents. Utilities have implemented both NRC-mandated changes and additional improvements on their own initiative. However, the changes were formed on the basis of the constraints associated with backfits to existing control rooms (CRs) using early 1980s technology, which limited the scope of corrective actions that might have been considered that is, more effective changes can be made in a new CR with the modern technology typical of advanced CRs). Problems and issues encountered in similar systems of previous designs will be identified and analyzed so that they are avoided in the development of the current system or, in the case of positive features, to ensure their retention.

Element 3 System Functional Requirements Analysis

System requirements shall be analyzed to identify functions that must be performed to satisfy the objectives of each functional area. System function analysis shall (1) determine the objectives, performance requirements, and constraints of the design, and (2) establish the functions that must be accomplished to meet these objectives and requirements.

Element 4 Allocation of Function

Functions shall be allocated to take advantage of human strengths and avoid the effects of human limitations. To ensure that function allocation is conducted according to accepted HFE principles, a structured and well-documented methodology of allocating functions to personnel, system elements, and personnel-system combinations shall be developed.

Element 5 Task Analysis

Task analysis shall provide the systematic study of the behavioral requirements of the tasks the personnel subsystem required to perform in order to achieve the allocated functions. The task analysis shall fulfill the following objectives:

- Provide one of the bases for making design decisions (for example, determining before hardware fabrication, to the extent practicable, whether system performance requirements can be met by combinations of anticipated equipment, software, and personnel).

- Ensure that human performance requirements do not exceed human capabilities.

- Be used as basic information for developing procedures, staffing, skill, training, and communication requirements of the system.

- Form the basis for specifying the requirements for the displays, data processing, and controls needed to carry out tasks.

Element 6 Human/System Interface Design

Human engineering principles and criteria shall be applied along with all other design requirements to identify, select, and design the particular equipment to be operated, maintained, and controlled by plant personnel.

Element 7 Plant and Emergency Operating Procedure Development

Plant and emergency operating procedures (EOPs) shall be developed to support and guide human interaction with plant systems and to control plant-related events and activities. Human engineering principles and criteria as well as other design requirements shall be applied to develop technically accurate, comprehensive, explicit, easy to use, and validated procedures. The following types of procedures are covered in element 7:

- normal plant and system operations (including startup, power, and shutdown operations)
- abnormal and emergency operations
- alarm response

Element 8 Human Factors Verification and Validation (V&V)

Successful incorporation of HFE into the final HSI design and the acceptability of the resulting HSI, shall be thoroughly evaluated as an integrated system using HFE evaluation procedures, guidelines, standards, and principles.

18.2 Human Factors Engineering Program

The NRC HFE PRM for advanced evolutionary reactors specifies that a formal HFE program (Element 1) should be established to guide HFE activities. The staff's DSER review of the CESSAR-DC identified several issues related to HFE PRM Element 1. These were designated as DSER Issues 18.3.1-1, 18.3.2-1, and 18.3.5-1.

The HFE PRM was developed assuming that an HFE program plan would be developed at the beginning of the HFE effort. However, ABB-CE had already completed significant HFE analysis and design activities (before the HFE PRM was developed). It was therefore considered appropriate to modify some of the details of the HFE PRM Element 1 criteria to accommodate completed activities, as long as the substantive contributions of HFE activities to plant safety were not compromised (so that the intent of the HFE program elements is accomplished, even though some differences may exist between specific HFE PRM criteria and ABB-CE activities).

18.2.1 Objective

The objective of this review is to evaluate ABB-CE's efforts related to HFE PRM Element 1 Human Factors Engineering Program Management.

18.2.2 Methodology

18.2.2.1 Material Reviewed

The following ABB-CE documents were used in this review:

- CESSAR-DC Section 18.2, "Design Team Organization and Responsibilities."
- CESSAR-DC Section 18.4.2, "Human Factors Program Plan," and Reference 4 of Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 02, September 29, 1993), hereafter referred to as the HFPP.

18.2.2.2 Review Scope

This review focused on (1) resolution of DSER issues, and (2) evaluation of ABB-CE documents with respect to the topics and general criteria of the HFE PRM. Table 18.2 provides a "compliance matrix," which includes a cross-reference between review items and the pertinent sections of the HFPP. As indicated in the introduction, absolute adherence to the HFE PRM was not considered mandatory. Differences in approach would be acceptable, provided that (1) the program can still meet the HFE commitment and goals, (2) the difference between the proposed criteria and those contained in the HFE PRM are adequately justified, and (3) there is no adverse impact on other program elements.

The System 80+ plant and CR design is quite far along, and there is a considerable amount of actual design material available. ABB-CE has included such design information along with the HFE program description. Hence, the scope of the documents reviewed goes beyond the "submittal requirements" of the HFE PRM. Because the review was model driven, those portions of the documents that were within the scope of the HFE PRM were reviewed. Portions of the HFPP that address design feature justification (HFPP Section 3, for example) were not reviewed.

Table 18.2 Comparison of HFE PRM and the ABB-CE HFPP

NRC Review Model Component	FSER	ABB-CE Plan Section
Purpose, Scope, and Organization	18.2.3.2.1	1.1, Appendix A
Goals and Objectives	18.2.3.2.2	1.2, 3, Appendix A
Management and Organization	18.2.3.2.3	
Design Team and Organization	18.2.3.2.3.1	1.3.1
Integration into Process	18.2.3.2.3.2	1.3
Program Milestones	18.2.3.2.3.3	1.3.1.2, 7
Documentation	18.2.3.2.3.4	1.3.2, 7
Subcontractor Efforts	18.2.3.2.3.5	1.3.1.4
Literature/Practices Review	18.2.3.2.3.6	Appendix A
Issue Tracking System	18.2.3.2.3.7	Appendix A
Technical Program	18.2.3.2.4	2 to 6, Appendix A

18.2.2.3 Review Procedure

The review began following the identification of DSER issues. A draft HFPP responding to DSER issues related to HFE PRM Element 1 was submitted to the review team in October 1992. A draft HFPP evaluation providing preliminary questions and raising points of clarification was prepared in November 1992. A meeting was then held with ABB-CE in November 1992, to discuss these review comments, and a telephone conference in early December clarified the reviewers' comments. The HFPP was revised following these discussions, and the review of the revised HFPP is the subject of this FSER.

The following materials were consulted as part of the evaluation:

- HFE PRM, forwarded to the Commission in SECY-92-299, dated August 27, 1992.
- Public meeting minutes from September 10 and 11, 1992, meetings between NRC and ABB-CE.
- Technical Evaluation Report, "System 80 Operating Experience Issues Based Upon Interviews with System 80 Operators," BNL Technical Report E2090-T2-4-3/93, J. O'Hara and W. Luckas, Jr., March 29, 1993 (BNL TER).
- U.S. Nuclear Regulatory Commission (1992), "Draft Safety Evaluation Report" (NUREG-1492), Washington, D.C.

18.2.3 Results

18.2.3.1 DSER Review

18.2.3.1.1 DSER Issues

The staff's initial review of this element identified the following DSER issues:

- 18.3.1-1 Human Factors Engineering Program
- 18.3.2-1 HFE Program Milestones and Task Schedules
- 18.3.5-1 Design Goals

At the public meeting on September 10 and 11, 1992, ABB-CE agreed to address the DSER issues by developing an HFE HFPP which fulfilled the following requirements:

- Address the human-centered design goals of HFE PRM criterion 1,and specify how the goals will be evaluated throughout the design process, including the V&V effort.
- Provide in the plan a schedule for tests and evaluation that (1) shows the relationship between the HF activities and the overall plant design process, and identifies the HFE products associated with the milestones.

18.2.3.1.2 DSER Issue Resolution

Item a: Address the human-centered design goals of HFE PRM criterion 1, and specify how the goals will be evaluated throughout the design process, including the V&V effort.

Evaluation: This item is resolved in Section 18.2.3.2.2, "Overall HFE Program Goals and Objectives," below.

Item b: Provide in the plan a schedule for tests and evaluation that (1) shows the relationship between the HF activities and the overall plant design process, and identifies the HFE products associated with the milestones.

Evaluation: This item is resolved in Section 18.2.3.2.2, "HFE Program Milestones," below.

18.2.3.2 HFE PRM Criteria-Based Evaluation

18.2.3.2.1 General Purpose, Scope, and Organization

Criterion: The HFE PRM specifies that the HFPP should address the overall purpose and organization of the HFPP.

Evaluation:

1. Purpose - The plan generally encompasses the topics identified in the HFE PRM and provides additional information by including overviews of completed analyses and design justification and bases for proposed design features. The additional information is not reviewed in this section (but will be reviewed in conjunction with the appropriate HFE PRM element reviews in the remaining sections of this chapter which address the technical details of the HFE program).

2. Scope - There is a one-to-one relationship between the main body of the plan description and the program element descriptions in Appendix A of the HFPP. The main body of the plan contains a description of the completed HFE activities to date and plans for future HFE activities performed as the design proceeds. Appendix A provides detailed goals, requirements, and criteria for these future HFE activities.

   A formal procedure element is explicitly excluded from the plan. ABB-CE states that detailed procedure development is a licensee activity (as described in CESSAR-DC Section 13.5). ABB-CE will prepare procedure guidelines as technical input to the licensee. Because this guidance will focus on content rather than format, it is not covered in the HFE program. Therefore, there is no element in the ABB-CE requirements document that corresponds to HFE PRM Element 7. The acceptability of this omission is evaluated in FSER Section 18.2.3.2.4.

3. Organization - The HFPP describes the HFE design team and the management structure for the HFE effort. The plan also provides a clear description of the overall technical program and its relationship to completed activities and planned analyses. No issues concerning HFPP organization were identified.

The applicant's plan acceptably addresses the overall purpose and organization of the plan. Hence, this criterion is satisfied.

18.2.3.2.2 Overall HFE Program Goals and Objectives

Criterion: The primary goal of the HFE program is to develop an HSI that makes possible safe, efficient, and reliable operator performance, and that satisfies all regulatory requirements stated in 10 CFR. The general goals of this program are "human-centered." As the HFE program develops, these goals will be objectively defined and shall serve as criteria for testing and evaluation activities. Generic "human-centered" HFE design goals are listed in General Criterion 1 of the HFE PRM.
Evaluation: Further discussion of the general goals was a provision of the public meeting on September 10 and 11, 1992, concerning DSER issue resolution (see Item "a" under Section 18.2.3.1.2, "DSER Issue Resolution," above). The design philosophy and subsidiary "philosophies" appear to reflect a reasonable and acceptable set of high-level design goals that are generally consistent with the HFE PRM.

The staff concludes that the HFPP is acceptable and establishes the commitment to evaluate human-centered goals as part of ABB-CE's HFE evaluations. Hence, this criterion is satisfied.

18.2.3.2.3 Program Management and Organization

18.2.3.2.3.1 HSI Design Team

Criteria: The HFE PRM provides criteria related to the organizational responsibility and expertise of the HFE design team which conducts the HFE program. The HSI design team should have the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is achieved.
The team should be responsible for such activities as developing HFE plans and procedures; the oversight and review of all HFE design, development, test, and evaluation efforts; initiating, recommending, and solving HFE problems identified during the design and implementation of human/system interfaces; verifying the implementation of team recommendations; assuring that HFE activities comply with HFE plans and procedures; and scheduling activities and milestones. The team should have the authority and organizational freedom to ensure that it controls its areas of responsibility and is able to identify problems in the implementation of the HSI design.

The HSI design team should include, at a minimum, the following areas of expertise: technical project management, systems engineering, nuclear engineering, instrument and control engineering, architectural engineering, human factors engineering, nuclear power plant operations, computer systems/software engineering, nuclear power plant procedures development, training program development, system safety engineering, and reliability, availability, maintainability, and inspectability (RAMI) engineering.

Evaluation: The HSI design team is described in CESSAR Section 18.2, "Design Team Organization and Responsibilities." The applicant's description of the HSI design team complies with the criteria stated above with the exceptions discussed below.

In CESSAR Section 18.2.2, "Nuplex 80+ Design Review Team," the applicant explains how HFE design decisions are made through design review meetings. However, the applicant did not adequately described the process that management uses to make decisions on HFE issues. Also missing was an explanation of how the HSI Design Team was involved in the decision-making process. The tools and techniques (e.g., review forms, project review meetings, documentation) the HSI Design Team uses to carry out their responsibilities was not been included as part of the description of the HSI Design Team with the exception of an explanation of "boiler room meetings."

The applicant's letter of July 31, 1992, "System 80 Human Factors Engineering team description and markup of Part II Human Factors Criteria," described the process that management uses to make decisions regarding HFE issues indicating that the majority of decisions are made at the technical level and resolved through review and consensus at review meetings. Decisions that cannot be resolved are brought to the attention of management for them to resolve. Further, the applicant indicated that an external design review team reviews design developments and the work of the HSI Design Team. Specifically, the external team reviews documents and the results of meetings produced by the Team. This process acceptably addresses the staff's concern.

The applicant's letter of July 31, 1992, "System 80 Human Factors Engineering team description and markup of Part II Human Factors Criteria," indicated that the HSI Design Team uses project documents as the primary tool to accomplish their work. These include plans, system descriptions, human factors standards and guidelines, verification reports, task analysis reports, and panel design reports. The applicant reported that other tools include design review meetings results that are documented through internal memoranda and the computerized tracking of open issues system which includes human factors efforts. The staff finds the information provided by ABB-CE regarding tools and techniques used by the HSI Design Team is acceptable.

In CESSAR Section 18.2, "Design Team Organization and Responsibilities," and Paragraph 1.2.1, "Organization of Design Team," of the Human Factors Engineering Program Plan of February 21, 1992, "Description of Human Factors Program for the System 80 standard plant design," the applicant describes the composition and general qualifications of the HSI Design Team. However, the applicant had not provided job descriptions of the team's members nor identified key personnel and their qualifications related to the their areas of expertise and their responsibilities on the team. In addition, it was not clear how the following expertise was integrated into the team: systems engineering, architectural engineering, nuclear power plant procedures development, training, systems safety engineering, and RAMI engineering.

The applicant's letter of July 31, 1992, "System 80 Human Factors Engineering team description and markup of Part II Human Factors Criteria," provided job descriptions for the following HSI Design Team members: Manager, Advanced Reactor Instrumentation and Control (I&C); Technical Supervisor, Control Complex Engineering; Consulting Engineer, HFE (I&C Department); Lead Engineer, HFE (Services Department); Senior Engineer, HFE (Services Department); Consulting Engineer, I&C; Lead Engineer, I&C/HF; Lead Engineer, I&C/HF/Operations; Consulting Engineer, I&C/Operations; Technical Supervisor, I&C; Consulting Engineer, I&C; and A/E Liaison and Operations Expert, Duke Engineering and Services. The applicant has satisfactorily described the jobs of the HSI Design Team members, and identified key personnel and their qualifications regarding their areas of expertise and responsibilities on the Team.

In the letter of July 31, 1992, the application indicated that the HSI Design Team possesses the following technical expertise; systems engineering, architectural engineering, nuclear power plant procedures development, personnel training/systems approach to training, systems safety engineering, and RAMI engineering. Previously, in CESSAR Section 18.2.2 and the letter of February 21, 1992, the applicant documented the Team possesses the following areas of expertise: technical project management, nuclear engineering, I&C engineering, HF engineering, nuclear power plant operations and computer systems/software engineering. Based on a review of these descriptions, the staff finds that the applicant has the appropriate expertise.

The criteria related to the HFE Design Team are satisfied.

18.2.3.2.3.2 Integration of HFE and Other Plant Design Activities

Criterion: According to the HFE PRM, the HFPP should identify integrated design activities.

Evaluation: Based upon the HFPP description, HFE is well integrated into the design process. There are three mechanisms for this integration. First, the design process shows HFE analyses and evaluations throughout the HSI design cycle. Figures 1.3-1 through 1.3-6 of the HFPP illustrate the relationship between HFE structured analyses and the system design. Second, there has been active involvement of HFE specialists in the multi-disciplinary design team. Third, HFE activities are part of design review meetings. Hence, this criterion is satisfied.

18.2.3.2.3.3 HFE Program Milestones

Criterion: According to the HFE PRM, the HFPP should identify HFE milestones and provide a program schedule.

Evaluation: Further discussion of the schedule and milestones was a provision of the public meeting on September 10 and 11, 1992, DSER issue resolution (see Item "b" under Section 18.2.3.1.2 "DSER Issue Resolution," above).

Generic HFE activities are described in Figure(s) 1.3-2 through 1.3-6 of the HFPP, which depict the flow of HFE efforts in terms of parallel and serial activities and interdependencies. Milestones and general documentation outputs are also illustrated in these figures. These figures provide a clear overview of the overall program and its products. Hence, this criterion is satisfied.

18.2.3.2.3.4 HFE Documentation

Criterion: According to the HFE PRM, the following items were expected for each element:

- Implementation Plan
- Analysis Report
- Design Team Review Report

Evaluation: Section 1.3.2 of the HFPP provides a list of HFE products, but several items specified by the HFE PRM are not on the list:

- Implementation plans for Elements 2 through 4. This is acceptable because ABB-CE's efforts in these areas are well underway (or in some cases essentially complete). Since the purpose of an implementation plan review is to review methodology, CE incorporated a description of their detailed methodology for conducting Element 2 to 4 in their submittals for those elements. These submittals were reviewed by the staff in FSER Sections 18.3, 18.4, and 18.5 respectively.

- Procedure development reports. This is addressed in Section 18.7 of this report.

- All reports of the design review evaluations. The reviewers were unsure of the function of the design review meetings (DRM) with respect to the HFE PRM. DRMs are an important aspect of any design effort and are commonly applied by most, if not all, developers of complex systems. In ABB-CE's HFPP, the DRM are also intended as the approach to satisfy design team reviews of technical HFE program products (as identified in the HFE PRM). The reviewers did not think that the approach meets the HFE PRM requirements of formal HFE product reviews or design review reports. At the public meeting between ABB-CE and the staff on November 19, 1992, ABB-CE stated that a document review process is used but not formally documented. ABB-CE agreed to provide additional information about the HFE product review process. The documentation of these reviews was requested to ensure the interdisciplinary review of all HFE efforts, and was provided on pages A-18 and A-19 of the HFPP. The plan states that "analysis reports will be subject to a formal interdisciplinary review and comment resolution process." This acceptably addresses the staff's concern, therefore, this criterion is satisfied.

18.2.3.2.3.5 HFE in Subcontractor Efforts

Criterion: The HFE PRM specifies that HFE in subcontractor efforts should fulfill the following requirements:

- Provide a copy of the HFE requirements proposed for inclusion in each subcontract.

- Describe the manner in which the designer proposes to monitor the subcontractor's compliance with HFE requirements.

Evaluation: In CESSAR, Section 18.2.1, "Nuplex 80+ Design Team," and in the Human Factors Engineering Program Plan of February 21, 1992, the applicant indicates in Paragraph 1.2.1.3, "Human Factors Efforts by Subcontractors," that Duke Engineering and Services (DE&S) is a subcontractor for "some balance of plant work relating to the man-machine interface for System 80+..." The staff's determined that HFE input to the work performed by DE&S occurs after, rather than during, the actual panel design and is performed as a review function by ABB-CE HFE, specialists, although ABB-CE stated that it retains final design authority, review, and responsibility for the work performed by Duke. The staff believes that human factors should be integrated throughout the production process used by subcontractors and not only for review as project milestones are achieved. The staff, therefore, found that the applicant's approach was not appropriate for future human factors efforts by subcontractors. The staff requested that the applicant (1) reconsider its present approach to reviewing products produced by subcontractors or (2) provide a justification that ensures subsequent products will reflect the full application of human factors in the design.

The applicant noted in its submittal of May 22, 1992, that the HFE Standards are part of the Human Factors Program Plan for System 80+. Further, the applicant stated in Section 1.2, "Applicability" of the HFE Standards of May 22, 1992: "The contents of the HFE Standards and Guidelines Document apply, as appropriate, to all System 80+ system and equipment designs built by ABB Combustion Engineering and its subcontractors..."

The staff finds that this information satisfactorily responds to the staff's concerns regarding HFE input to work by subcontractors. Hence, this criterion is satisfied.

18.2.3.2.3.6 Literature and Current Practices Review

Criterion: HFE PRM General Criterion 2 identifies acceptable references upon which an HFE program can be developed.

Evaluation: HFPP Appendix A indicates that the HFPP and related criteria were based upon a review of 15 source documents. These include many of the same sources used as technical bases of the HFE PRM. Hence, this criterion is satisfied.

18.2.3.2.3.7 HFE Issues Tracking System

Criterion: The HFE PRM identifies an human factors issues tracking system. The method used for the tracking system should document and track human factors engineering issues and concerns from the time they are identified until they have been eliminated or reduced to a level acceptable to the applicant's multidisciplinary review team. Each issue identified to qualify for tracking should be documented along with the action taken to reduce or eliminate the issue/concern, and the final resolution should also be documented in detail (e.g., person accepting, date).

Evaluation: In a letter of May 8, 1992, the applicant identified an HFE issues tracking system as a part of the Human Factors Engineering Program Management Plan. In its submittal, the applicant described an I&C department comment-resolution tracking system that is used to assure future implementation of open items identified during the design process. The applicant agreed to include a dedicated segment for tracking human factors issues; the system will become a long-term, full-scope tracking method. The applicant has described a tracking system that satisfies the criterion for a human factors issues tracking system. This system was implemented and entries in it were verified by the staff for adherence to the criteria. Hence, this criterion is satisfied.

18.2.3.2.4 Technical Program

Criterion: Identify and describe the development of implementation plans, analyses, and the evaluation and verification of

- operating experience review
- system functional requirements development
- allocation of function
- task analysis
- interface design
- plant and emergency operating procedure development
- HF verification and validation

Evaluation: ABB-CE's HFE program is organized into the following eight components:

- Element 1 human factors engineering program management
- Element 2 incorporation of industry experience
- Element 3 evaluation and allocation of system functions
- Element 4 task analysis
- Element 5 man-machine interface design
- Element 6 availability verification
- Element 7 suitability verification
- Element 8 validation of ensemble.

Table 18.3 shows the general relationship between the HFE PRM and ABB-CE model components. ABB-CE's technical program described in the HFPP is reviewed using HFE PRM criterion to verify the incorporation of essential HFE elements. A detailed review of each technical HFE element is provided in the remaining sections of this FSER. Table 18.3 shows the FSER section in which the technical details of each element are reviewed.

The ABB-CE HFE technical program plan acceptably contains all but one of the main components of the HFE PRM. The absence of a procedures element component is discussed below.

Procedure development was defined in the HFE PRM as one of eight fundamental elements of an HFE program. Along with other HFE program elements, procedure

Table 18.3 Relationship between the HFE PRM and ABB-CE HFPP components

PRM	HFPP	STATUS	HFE FSER SECTION
1	1	Generally consistent with the HFE PRM	18.2
2	2	Generally consistent with the HFE PRM	18.3
3	3*	Generally consistent with the HFE PRM	18.4
4	3*	Generally consistent with the HFE PRM	18.4
5	4	Generally consistent with the HFE PRM	18.5
6	5	Generally consistent with the HFE PRM	18.6
7	-	See Procedure Element discussion	18.7
8	6**	Generally consistent with the HFE PRM	18.8
	7**	Generally consistent with the HFE PRM	18.8
	8**	Generally consistent with the HFE PRM	18.8

* ABB-CE HFPP combines function requirements and allocation.
** ABB-CE HFPP addresses V&V in components 6, 7, & 8.

development contributes to the successful integration of plant personnel and systems, thereby supporting public health and safety. ABB-CE has not included detailed procedure development as part of the System 80+ HFE program.

This oversight raised two concerns related to (1) the incorporation of procedure development into the HFE development process to ensure procedures that reflected HFE considerations, and (2) system validation with the final design including procedures. Both concerns have been resolved. The detailed evaluation of these resolutions can be found in FSER Sections 18.7 and 18.8, respectively. To summarize, ABB-CE has committed to include procedure development as COL Action Item 13.5.1, Plant Operating Procedure (POP) Development Plan in the CESSAR-DC. ABB-CE will develop the technical information required to serve as a basis for detailed procedure development as part of the HFE process, and this information will be provided to the COL applicant. This COL action item is acceptable.

With respect to impact on validation, ABB-CE included in the CESSAR-DC, a requirement in COL Action Item 13.5.1, POP Development Plan, for the COL applicant or holder to perform a POP validation effort that demonstrates the acceptability of the completed procedures. CESSAR-DC Section 18.9.3, "Validation," was then modified to break validation into two phases. Section 18.9.3.1, Design Validation, addresses validation of the entire HSI without final procedures. CESSAR-DC Section 18.9.3.2, "Operating Ensemble Validation Plan," and HFE V&V Plan Section 6.3.4.4, Operating Ensemble Validation Activities, address the "final" validation of the HSI after the final procedures have been completed. Operating ensemble validation requirements are addressed in CESSAR-DC Section 18.9.3.2. This validation, which will be performed by the COL applicant, will ensure that trained operators using final, plant-specific procedures in the as-built CR form an effective operating ensemble , this two-phased validation approach and associated COL action item are found acceptable, therefore, this criterion is satisfied.

18.2.4 Findings

The ABB-CE HFPP and related sections of the CESSAR-DC acceptably address the requirements of HFE PRM Element 1, "Human Factors Engineering Program Management." While the HFPP did not include procedure development as part of its technical program, ABB-CE has modified the CESSAR-DC to incorporate a COL action item to address aspects of procedure development that were required by the HFE PRM but not addressed in the ABB-CE's HF program. Therefore, the criteria of HFE PRM Element 1 are acceptably met, and this COL action item is acceptable.

18.3 Operating Experience Review

The HFE PRM specified that an OER (Element 2) should be performed. The staff's DSER review of the CESSAR-DC identified OER as DSER Issue 18.4. After the DSER was issued, ABB-CE issued its OER in December 1992. The staff reviewed this document and identified a number of issues. These were resolved during an iterative review process, and a revised OER was issued in June 1993.

18.3.1 Objectives

The objective of the OER evaluation was to assess ABB-CE's efforts related to HFE PRM Element 2, "Operating Experience Review."

18.3.2 Methodology

18.3.2.1 Material Reviewed

The following ABB-CE documents were used in this evaluation:

- Reference 1 of CESSAR-DC Section 18.4, "Operating Experience Review for System 80+ MMI Design" (NPX80-IC-RR790-01, Rev. 01, June 9, 1993).
- Reference 1 of CESSAR-DC Section 18.6, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).
- Reference 4 of CESSAR-DC Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 02, September 29, 1993).
- CESSAR-DC Chapter 17, Appendix A, "Closure of Unresolved and Generic Safety Issues through Amendment Q," June 30, 1993.
- Reference 1 of CESSAR-DC Section 18.10, LD-92-076, "System 80+ Shutdown Risk Report, Revision 1," Attachment "System 80+ Shutdown Risk Evaluation Report" (DCTR 10, Draft, June 15, 1992), ABB-CE letter dated June 16, 1992.
- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 2, "Nuplex 80+ Compliance with NUREG-0737 Supplement 1 Requirements," ABB-CE letter dated September 23, 1993.
- Reference 2 of CESSAR-DC Section 18.10, LD-92-115, "Closure of System 80+ Draft Safety Evaluation Report Issues," attached response to DSER Issue No. 20.2-28, ABB-CE letter dated November 24, 1992.
- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), attached response to DSER Item 20.2-23 and 20.2-29, ABB-CE letter dated December 18, 1992.
- Reference 4 of CESSAR-DC Section 18.10, LD-93-135, "System 80+ Information for Issue Closure," Attachment 1, "ABB-CE Response to System 80 Operating Experience Issues Based Upon Interviews with System 80 Operators," ABB-CE letter dated September 1, 1993.
- Reference 5 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 5, "CESSAR-DC Markups for V&V and Procedures," ABB-CE letter dated September 24, 1993.

18.3.2.2 Review Scope

This review focused on (1) the overall scope, structure, and completeness of the ABB-CE documents, and (2) the evaluation of the documents with respect to the HFE PRM. In conducting this review, absolute adherence to the HFE PRM was not considered mandatory. Differences in approach were considered acceptable provided (1) the program could still meet the HFE commitment and goals, (2) the difference between the proposed criteria and those contained in the HFE PRM were adequately justified, and (3) there was no adverse impact on other program elements.

18.3.2.3 Review Procedure

The OER was reviewed using the HFE PRM. Further, since the OER addresses various NRC unresolved and generic safety issues, a number of ABB-CE and NRC documents covering these items were also reviewed. The unresolved and generic issues were reviewed for the satisfactory resolution of their HFE aspects.

A concern identified early in the review related to the adequacy of ABB-CE's review of operating experience relevant to System 80, the immediate predecessor of the System 80+ plant. As a result, the reviewers collected and reviewed licensee event reports (LERs) from System 80 plants, and visited a System 80 plant and interviewed operators regarding their opinions of the HFE and their plant operating experience.

The following materials were consulted as part of the evaluation:

- HFE PRM, forwarded to the Commission in SECY-92-299, dated August 27, 1992.
- International Electrotechnical Commission (1989), "International Standard: Design for Control Rooms of

Nuclear Power Plants" (IEC-964), Geneva, Switzerland: Bureau Central de la Commission Electrotechnique Internationale.

- LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment "Control Complex Information System Bases for Nuplex 80+" (NPX80-IC-DB791-01, January 15, 1993), ABB-CE letter dated January 18, 1993.
- Reference 1 of CESSAR-DC Section 18.10, LD-92-076, "System 80+ Shutdown Risk Report, Revision 1," Attachment, "System 80+ Shutdown Risk Evaluation Report" (DCTR 10, Draft, June 15, 1992), ABB-CE letter dated June 16, 1992.
- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX80-IC-DP-790-01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.
- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Operating Experience Review for System 80+ MMI Design" (NPX80-IC-RR790-01, Rev. 00), ABB-CE letter dated December 18, 1992.
- Reference 7 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 01, December 8, 1992), ABB-CE letter dated December 18, 1992.
- NRC Internal Memorandum, Request for HICB Review of System 80+ Design Features I & C, (J.S. Wermiel to W. Swenson), June 16, 1993.
- NRC Internal Memorandum, Request for HICB Review of System 80+ Design Features I & C, (J.S. Wermiel to W. Swenson), June 23, 1993.
- Nuclear Management and Resources Council (1991), "Guidelines for Industry Actions to Assess Shutdown Management," (NUMARC 92-106), Washington, D.C.
- Nuclear Safety Analysis Center (1981), "Verification and Validation for Safety Parameter Display Systems" (NSAC-39), Palo Alto, CA.
- Public meeting minutes from September 10 and 11, 1992, meeting between NRC and ABB-CE.
- Public meeting minutes from April 19 through 21, 1993, meeting between NRC and ABB-CE.
- Public meeting minutes from May 13 and 14, 1993, meeting between NRC and ABB-CE.
- Technical Evaluation Report, "Review of the System 80+ Operating Experience Review," BNL Technical Report E2090-T2-5-3/93, J. Higgins and J. O'Hara, March 31, 1993 (BNL TER).
- U.S. Nuclear Regulatory Commission (1992), "Draft Safety Evaluation Report" (NUREG-1492), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1980, 1982), "Clarification of TMI Action Plan Requirements" (NUREG-0737 and Supplement 1), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1988), "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems" (NUREG-1342), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1989), "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit" (Regulatory Guide (RG) 1.114, Rev. 02), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1991), "Resolution of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, Pursuant to 10 CFR 50.54(f)" (Generic Letter 91-06), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1991), "Reactor Coolant Pump Seal Failures and Its Possible Effect on Station Blackout" (Generic Letter 91-07), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1991), "Resolution of Generic Issues 48, LCOs for Class 1E Vital Instrument Buses, and 49, Interlocks and LCOs for Class 1E Type Breakers Pursuant to 10 CFR 50.54(f)" (Generic Letter 91-11), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1992), "A Prioritization of Generic Safety Issues" (NUREG-0933), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1992), "Standard Technical Specifications Combustion Engineering Plants" (NUREG-1432), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1993), "Shutdown and Low-Power Operation at Nuclear Power Plants in the United States" (NUREG-1449), Washington, D.C.

18.3.3 Results

18.3.3.1 DSER Issues Review

18.3.3.1.1 DSER Issue

In the staff's initial review of this element, DSER Issue 18-4 was defined, indicating that ABB-CE had not submitted an OER.

18.3.3.1.2 Issue Resolution

At the public meeting on September 10 and 11, 1992, ABB-CE agreed to address the DSER issue by

- identifying past problems and lessons learned (in an organized, coordinated, usable, and auditable form) for the CR, remote shutdown panel, and local control stations

- giving examples and the rationale for problems and issues in similar systems of previous designs to identify and analyze negative features and retain positive features

- addressing the criteria of HFE PRM Element 2

- submitting the System 80+ Design Basis Document and System 80+ Information System Design Basis Document

Evaluation: The ABB-CE OER (Rev. 00) and the revised OER (Rev. 01) contain the information identified in the first three items, above. The adequacy of the OER submittal is discussed below. The System 80+ Design Basis Document and System 80+ Information System Design Basis Document were submitted to the NRC and were used for the review of other HFE PRM elements. This issue is resolved because CE has now submitted an OER.

18.3.3.2 HFE PRM Criteria-Based Evaluation

The initial ABB-CE OER and the revised OER were evaluated according to the criteria of the HFE PRM. The results are discussed below.

18.3.3.2.1 Implementation Plan

Criterion: An OER implementation plan shall be developed.

Evaluation: As per the review of the ABB-CE HFPP (Reference 4 of CESSAR-DC Section 18.4), implementation plans are not required for HFE program elements currently underway or completed. Instead, a description of the methodology used is to be incorporated in the report. ABB-CE describes their OER process in Sections 1, 2, and 5.2 of the OER document. This satisfies the need to document the OER scope and process for System 80+; hence, a separate implementation plan is not needed.

OER is generally comprehensive in its scope and level of detail. The OER states that guidance and associated design resolutions apply to the entire System 80+ design, which is considered appropriate. It also states that all areas of the plant are being subjected to a detailed operability and maintainability review. This is considered acceptable design practice. Further, the commitment to continue to review new industry and government reports and other applicable documents is considered an acceptable practice.

However, the implementation scope of the initial OER was too limited in the following areas:

- remote shutdown panels
- local control stations
- review of System 80 experience

The staff noted that the OER should address recent documents on local control stations developed in the review of the HF generic issue on local control stations, and those documents noted in paragraph 18.3.3.2.6, below. A list of seven pertinent local control station documents was provided to ABB-CE by the staff. ABB-CE reviewed these and documented the review in Appendix C of the revised OER. Design guidance was some items in the System 80+. These issues were entered into the HFE tracking system. Further, the ABB-CE OER design resolutions appeared to somewhat narrowly exclude local control stations and the remote shutdown panel. The revised OER has modified the design resolution section of these items to include local control stations within their scope. Therefore the PRM criterion is satisfied. The review of System 80 experience will be discussed 18.3.3.-2.7, below.

18.3.3.2.2 Analysis Results Report

Criterion: The analysis of operating experience shall be conducted in accordance with the plan, with the findings documented in an evaluation report.

Evaluation: The OER is the evaluation report for this element of the HFE PRM. This report contains the objectives, methods, results, conclusions, and recommendations or implications for HSI design of the OER as required by the HFE PRM and satisfies this criterion.

Section 3 of the ABB-CE OER contains the detailed results of the OER analysis, and addresses a considerable number of HF/HSI issues. ABB-CE modified the OER based on the staff's review of the initial OER. The following modifications are particularly noteworthy:

- Section 3.6.1, "Inconsistent Coding Conventions," also applies to local control stations.
- Section 3.6.2, "Insufficient Tag Legibility," is clarified to say that it applies to the main control room (MCR), the remote shutdown panel, and other local control stations.
- Section 3.8.2, "Standardization of Man-Machine Interface" (MMI), also applies to remote shutdown panels and LCSs.

18.3.3.2.3 HSI Design Team Report Review

Criterion: The analysis shall be reviewed by the HSI design team and shall be documented in an evaluation report.

Evaluation: ABB-CE did not initially provide in the HFPP a description of a formalized design team review of the final analysis reports of each process element. At the NRC meeting of November 19, 1992, ABB-CE stated that it performs an interdisciplinary design team review of each of the major design element results, and that this review is formally documented. Although ABB-CE described its design review process in HFPP Section 1.3.1.3, "Design Review Meetings," it did not meet the HFE PRM requirements for HSI design team evaluation. ABB-CE subsequently revised the HFPP to address the concern, and an open item related to design team review was resolved. Additionally, OER Section 5.4 summarizes the design team review of the OER. Therefore, this criterion is satisfied.

18.3.3.2.4 Issues Identification (Appendix A)

Criterion: As part of the design and implementation process, the OER should include the issues listed in Appendix A of the HFE PRM

Evaluation: In the ABB-CE OER, Appendix A discusses the list of issues from the HFE PRM, including all of the types of issues documents: unresolved safety issues (USIs), TMI issues, NRC generic letters, Office for Analysis and Evaluation of Operational Data (AEOD) studies, and low-power and shutdown issues. All of the USIs, TMI issues, and NRC generic letters listed in the HFE PRM were addressed in Appendix A of the OER. Appendix A also addressed AEOD studies and low-power and shutdown issues. Hence, this criterion is satisfied.

18.3.3.2.5 USI, GSI, and TMI Action Items

The staff's DSER indicated that several USI, generic safety issues (GSI), and TMI action items would be addressed in the FSER. Each of the issues related to this chapter is discussed below.

<u>DSER Issue 20.2-10: GSI Issue I.A.1.4 (Long-Term Upgrading of Operating Personnel and Staffing)</u>

Issue I.A.1.4 was considered by the staff to be beyond the scope of the design certification. The COL applicant will have responsibility for addressing this issue as part of the licensing process. In CESSAR-DC Chapter 1, ABB-CE identifies this issue as COL Action Item 20.2-10. This COL action item is acceptable.

<u>DSER Issue 20.2-11: GSI Issue I.C.9 (Long-Term Program for Upgrading Procedures)</u>

The staff reviewed GSI Issue I.C.9, "Long Term Plan for Upgrading Procedures," and determined that development of detailed procedures is beyond the scope of the System 80+ certification and will be the responsibility of the COL applicant. In CESSAR-DC Chapter 1, ABB-CE identifies this issue as COL Action Item 20.2-11. This COL action item is acceptable.

<u>DSER Issue 20.3-1: TMI Action Item I.A.4.2 (Simulator Capabilities) and Item II.J.3.1 (Management Plan for Design and Construction Activities)</u>

Sections 50.34(f)(2)(i) and 50.34(f)(3)(vii) correspond to TMI Action Items I.A.4.2 on simulator capabilities and II.J.3.1 on the management plan for construction activities, respectively. The latter item includes proposed procedures for handling the transition to operations. In CESSAR-DC Appendix A, ABB-CE identifies these issues as COL

Action Item 20.3-1. Therefore, the COL applicant will have responsibility for addressing these issues as part of the licensing process. This COL action item is acceptable.

DSER Issue 20.1-19: USI Issue B-17 (Criteria for Safety-Related Operator Actions-SROA)

This issue involved the development of a time criterion for safety-related operator actions including a determination of whether automatic actuation is required. Development and implementation of criteria for SROAs would likely result in the automation of some actions currently performed by operators. This should reduce the frequency of operator errors during transients or accidents. This issue also concerns some current pressurized water reactor (PWR) designs requiring manual operations to accomplish the switch-over from the injection mode to the recirculation mode following a loss-of-coolant accident (LOCA).

By agreeing to formalize the control room design process, including the use of Function Analysis, Functional Allocation (i.e. between operators and automated systems), and Task Analysis, CE has incorporated the concerns of this issue. Actions that are candidates for automation should be identified and a design that appropriately reduces the frequency of operator error should be produced. Further, ABB-CE indicated that the goal of the System 80+ design is that no manipulations requiring operator actions are required during the first 30 minutes for all System 80+ design-basis events.

Additionally, in the revised OER, ABB-CE indicates that the requirement for automation of the switchover from the injection mode to the recirculation mode is not applicable because the System 80+ design has an in-containment refueling water tank. ABB-CE notes that the System 80+ design eliminates the switch-over function. The staff finds the information provided by ABB-CE acceptably addresses the concerns of this issue. The specific example of switch-over to recirculation has been handled by a systems design change and the general aspect of the concern has been addressed by the HFE and Control room design process. Therefore, this issue is resolved.

DSER Issue 20.2-17: GSI Issue 125.I.3 (Safety Parameter Display System Availability)

This issue addressed SPDS availability and the reliability of the information it displays. In CESSAR-DC Appendix A, "Closure of Unresolved and Generic Safety Issues," ABB-CE states that the SPDS function will be performed by the advanced control complex comprised on the IPSO, DPS, and DIAS systems. Additionally each of these systems incorporates separate and redundant hardware, power supplies and self-test features. In Section 18.7.1.-8.2 of the CESSAR, ABB-CE indicated that the DPS, which provides the System 80+ SPDS function, has a reliability of greater than 99.99 percent.

In DSER Section 7.7.1.21, "Data Processing System," the staff indicated that one of the major functions of the DPS is validation of sensed parameters. In addition, the staff noted that ABB-CE states that the verification and validation DPS software modules are implemented in accordance with NSAC-39, "Verification and Validation for Safety Parameter Display Systems." As noted in NUREG-1342, this methodology provides some assurance that the SPDS software has been adequately designed, implemented, and tested.

As shown in CESSAR-DC (e.g., Figure 18.7.1-5), each DPS display page has in the upper right hand corner a dedicated space for the date and time which is continuously displayed. The time, provided in hour, minutes, and seconds, confirms for the operator whether the DPS system is active. In addition, validated parameters are displayed on DPS. In CESSAR-DC Appendix A, ABB-CE indicates that the DPS is configured redundantly for improved reliability. In addition, the DPS acquires and validates plant data. Additional discussion on information systems important to safety is discussed in Section 7.5 of this report.

The staff finds that the information reviewed above is acceptable because the design of the SPDS system is integrated directly into the IPSO, DPS, and DIAS systems which provide adequate separation and redundancy to ensure availability. Therefore, this issue is resolved.

DSER Issue 20.2-21: GSI Issue I.C.1 (Guidance for Evaluation and Development of Procedures for Transients and Accidents)

In the DSER, the staff indicated that ABB-CE should include a requirement that the owner-operator of a System 80+ plant provide plant-specific EOPs to comply with guidance in NUREG-0737 and its Supplement 1.

In Reference 5 of CESSAR-DC Section 18.10, LD-93-140, ABB-CE provides a markup of CESSAR-DC changes indicating that information concerning the site operator's plant procedures is within the site operator's scope and shall be provides in the site-specific safety analysis report. Further, ABB-CE provided a markup of a COL action item for procedures development, "Plant Operating Procedures Development Plan." The COL applicant will have responsibility for addressing this issue as part of the licensing process. In CESSAR-DC, Appendix A, ABB-CE has identified this issue as COL Action Item 13.5-1, "Site-

Specific Plant Operating Procedures." Furthermore, the area of procedural development is covered by appropriate ITAAC and its implementation will be reviewed by the NRC during the post-certification COL licensing process. This COL action item is therefore found to be acceptable.

DSER Issue 20.2-22: GSI Issue I.D.2 (Plant Safety Parameter Display System Console)

Issue I.D.2 in NUREG-0933 identified the need for the provision of an SPDS that displays a minimum set of parameters which define the safety status of the plant. In Section 18.7.1.8.1 of CESSAR-DC and the revised OER, ABB-CE indicates how System 80+ complies with the Supplement 1 to NUREG-0737 SPDS criteria. The staff reviewed the System 80+ advanced CR design against those criteria and found it acceptable. The results of this review are described in greater detail in Section 18.6.1.3.-1.4. The staff found that ABB-CE's responses and commitments regarding the eight SPDS requirements of Supplement 1 to NUREG-0737 are acceptable and, therefore, the DSER open item and GSI Issue I.D.2 are resolved.

DSER Issue 20.2-23: Issue I.D.4 (Control Room Design Standard)

Issue I.D.4 in NUREG-0933 addressed the need for guidance on the design of CRs to incorporate HF considerations. By letter dated December 18, 1992 (LD-92-120), ABB-CE indicated that this issue is resolved by (1) the System 80+ HF program which is being conducted in accordance with an HFPP for System 80+, that is based on current HFE program guidance, and (2) the System 80+ human factors engineering standards, guidelines, and bases. The staff finds this information acceptable and, therefore, this issue is resolved.

DSER Issue 20.2-24: GSI Issue I.D.5(1) (Control Room Design - Improved Instrumentation Research Alarms and Displays)

The DSER noted that this issue would be discussed in the FSER. Issue I.D.5(1) in NUREG-0933 involved the human-machine interface in the CR with regard to the use of lights, alarms, and annunciators to reduce the potential for operator error, information overload, unwanted distractions, and insufficient information organization.

ABB-CE has provided lighting and illumination levels in the "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (SGB document - Reference 1 of CESSAR-DC Section 18.6). Following the resolution of staff concerns regarding the SGB document (discussed in section 18.6.3 below), the technical adequacy of the aforementioned document was found acceptable. The staff evaluated the System 80+ annunciator and alarm systems during the onsite design features evaluation and concluded in the minutes of the public meeting held on May 13 and 14, 1993 these systems are acceptable except for the issues that were raised. The staff indicated in the May 13 and 14, 1993, public meeting minutes, that the issues could be resolved by ABB-CE's commitment to incorporate the issues into its HFE tracking system. ABB-CE will address the staff's specific concerns through evaluation and resolution of specific alarm system issues in its HFE tracking system. Issue 101, which provides a commitment for prototype testing, and a number of prior items that provide for continued tracking of the concerns raised during the meeting, have been included in the tracking system.

The staff finds that ABB-CE's information and commitments discussed above are acceptable and, therefore, this issue is resolved.

DSER Issue 20.2-27, GSI Issues I.D.3 and II.K.1.5 Regarding Safety-Related Valve Position Description

Issues II.K.1.5 and I.D.3 in NUREG-0933 addressed the direct position-indication of relief and safety valve position in the CR, such that the alarming and indication valve status should be clear and unambiguous and should be evaluated for HFE design considerations.

In CESSAR-DC Amendment Q (i.e., revised OER and CESSAR-DC Appendix A, "Closure of Unresolved and Generic Safety Issues"), ABB-CE indicated that the System 80+ CR has dedicated alarms to inform the operators when a valve has opened, providing unambiguous, direct indication of an open or partially open safety or relief valve. This information is acceptable and, therefore, GSI Issues I.D.3 and II.K.1.5 are resolved.

DSER Issue 20.2-28: GSI Issue II.K.1.10 (Review and Modify Procedures for Removing Safety-Related Systems from Service)

Issue II.K.1.10 in NUREG-0933 addressed the need to improve procedures. By letter dated November 24, 1992 (Reference 2 of CESSAR-DC Section 18.10, LD-92-115), ABB-CE indicated that this issue is not within the scope of design certification and has been made COL action item 13.14 as noted in CESSAR-DC Section 13.5.2, "Administrative Control Procedures." The staff agrees that this issue is not within the scope of design certification and finds the COL action item 13.14 to be acceptable. Therefore, this issue is resolved.

DSER Issue 20.2-29: Seven Human Factors-Related GSIs

(1) GSI Issue HF5.3 (MMI - Evaluation of Operational Aid Systems): By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120), ABB-CE provided information regarding this issue. ABB-CE indicated that the System 80+ MMI employs operator aids primarily to process data prior to presentation to the CR operators. The aids are integrated into the presentation hierarchy through application programs of the DPS and DIAS. The following operational aids are provided as part of the System 80+ MMI (with the corresponding CESSAR-DC sections indicated):

- signal reduction and validation, 18.7.1.4 and 18.7.3.2-.1.6
- integrated process status overview, 18.7.1.2
- alarm handling, 18.7.1.5
- critical function monitoring, 18.7.1.8.2 and 7.7.1.10
- success path monitoring, 18.7.1.8.2
- core limit monitoring, 7.7.1.8.1
- computer-aided surveillance testing, 7.7.1.8.2.M

The staff finds the above information acceptable and, therefore, this issue is resolved.

(2) GSI Issue HF4.5 (Application of Automation and Artificial Intelligence): By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120), ABB-CE indicated that critical function, success path functions and allocation (manual or automatic) have been retained from the predecessor System 80 design. Two functions (automatic closure of shutdown cooling system isolation valves and recirculation actuation) have been eliminated by design improvements. Automated functions or features include automatic load dispatch and margin preservation by the megawatt demand setter, validated aggregation of data, alarm mode dependency, explicit display of derived parameters, low-power feedwater control, automatic PPS surveillance, computer-aided testing (ESF) and success path monitoring. ABB-CE noted that there is no artificial intelligence used in System 80+.

The staff finds the above information acceptable and, therefore, this issue is resolved.

(3) GSI Issue HF5.4 (MMI - Computers and Computer Displays): This issue related to an evaluation of the safety significance and problems relating to the management of data and information in the control room during abnormal events. By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120), ABB-CE indicated that computer-based MMI are designed according to the HF program design process and meet the criteria of the System 80+ human factors engineering standards, guidelines, and bases. The staff finds this information acceptable, since the HF program design process will address the management of data and information during abnormal events; therefore, this issue is resolved.

(4) GSI Issue HF1.4.4 (Guidelines for Upgrading Other Procedures): The staff has reviewed HF1.4.4, "Guidelines for Upgrading Other Procedures," and determined that development of detailed procedures is beyond the scope of the System 80+ certification and will be the responsibility of the COL applicant. ABB-CE has included the procedure development process as COL Action Item 13.5-1, "Site-Specific Plant Operating Procedures," as described in CESSAR-DC Section 13.5. This COL action item is acceptable.

(5) GSI Issue HF5.1 (MMI - Local Control Stations): This issue notes that regulatory efforts dealing with human/systems interface had been limited to the control room and the remote shutdown panel and that further guidance is necessary regarding local control stations. By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120) and CESSAR-DC Appendix A, ABB-CE provided information regarding this issue. ABB-CE indicated that all System 80+ local control stations are designed in accordance with the criteria in the "Human Factors Engineering Standards, Guidelines, and Bases for System 80" (Reference 1 of CESSAR-DC Section 1.8.6). Further, ABB-CE noted that local control stations required to perform the System 80+ emergency operations guidelines are designed using task analysis and HF V&V. The staff finds this information acceptable and, therefore, this issue is resolved.

(6) GSI Issue HF5.2 (MMI - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation): This issue deals with the utilization of advanced technologies to improve control room annunciator systems and the fact that current guidelines do not address advanced technologies that are being introduced into new plant designs. By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120), ABB-CE conveyed that the System 80+ HF program is being conducted in accordance with the HF program plan for System 80+, described in CESSAR-

DC Section 18.4.2, and the "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (Reference 6 of CESSAR-DC Section 18.4). Further, ABB-CE noted that the NRC's draft NUREG-5908, "Advanced Control Room Design Review Guideline," was made available and used to evaluate the System 80+ advanced CR design.

ABB-CE does in fact plan to utilize such advanced concepts into their alarm system for the System 80+. ABB-CE describes the System 80+ annunciator design in CESSAR-DC Sections 18.7.1.4, "Nuplex 80+ Information Presentation" and 18.7.1.5, "Alarm Characteristics." In CESSAR-DC Section 18.7.1.5, ABB-CE indicates that priority 1, 2, and 3 alarms are processed and displayed independently through both the DIAS and the DPS. The staff reviewed the relevant design documentation and conducted an onsite evaluation of the mockup of the DIAS and DPS, including the annunciator alarm system. A discussion of the staff's onsite evaluation is provided in Section 18.6.1.2.1 of the FSER. Results of the staff's review of the System 80+ alarm system relevant to Issue HF1.3.4.b are provided below.

One characteristic of the DPS is as follows: The DPS display hierarchy provides access to displays incorporating system/component status, process parameters, and annunciator status/acknowledgement. ABB-CE demonstrated available portions of the DPS display hierarchy on the mockup, including display navigation paths based on plant CSFs and plant segments and the representation of process parameters and system/component status via DPS displays. Also demonstrated were the incorporation of alarm status representations into these displays and the alarm acknowledgement capability. The incorporation of the alarms into the plant displays provides the capability to access alarm condition information and then acknowledge alarms from any DPS CRT in the CR. This characteristic provides flexibility to control room operations. IEC 964 (1.4-1) states, "An alarm shall be annunciated in the CR section where the operator has the necessary means for initiating corrective actions." The System 80+ CR provides this capability in two ways (1) the DIAS has alarm display devices that are spatially dedicated to specific control panels where the relevant controls are located, and (2) the DPS displays can also be accessed from the relevant control panels. The staff found this DPS characteristic acceptable, as is further described in section 18.6.

The onsite review also examined the DIAS alarm tile displays that were resident on the RCS panel and the CVCS panel. The RCS panel DIAS alarm tiles contained groups of alarms that were functionally related to each other and to the RCS panel. The alarm tiles were spatially dedicated within the display page. The DIAS alarm tiles were presented on electro-luminescent panels on the vertical section of the RCS panel.

The DIAS alarm tile display system is coordinated with the DPS display system such that (1) the same coding schemes are used in the DIAS and DPS for indicating alarm priority and status, (2) the similar alarm messages appear in both the DIAS and DPS message windows (DPS messages are more detailed), and (3) alarms that are acknowledged by the operator on one system are also acknowledged on the other system.

The DIAS alarm tile display system is an operator alerting system that conveys the meaning and importance of alarm conditions through a hierarchical classification of alarm conditions and spatial dedication of alarm messages. This concept was found to be acceptable based on current alarm system guidelines and research addressing the value of alarm message prioritization/filtering and spatial dedication as techniques for reducing operator workload associated with handling alarm messages.

In summary, the staff finds that the documentation reviewed and the results of onsite evaluation relative to the System 80+ alarm system are acceptable and, therefore, this issue is resolved.

(7) <u>GSI Issue HF1.1 (Shift Staffing)</u>: By letter dated December 18, 1992 (LD-92-120), ABB-CE addressed GSI Issue HF1.1, "Shift Staffing." ABB-CE indicated that the System 80+ technical specifications (CESSAR-DC Chapter 16) identify minimum shift staffing requirements (Chapter 5) in accordance with the restructured Technical Specifications for ABB-CE plants (NUREG-1432). Further, ABB-CE noted that implementation of this requirement in accordance with RG 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit," will be documented as part of the site organization by the COL applicant. The staff interprets this requirement to be a part of COL Action Item 13.1-1. This COL action item is acceptable.

18.3.3.2.6 Review of HFE Issues

Criterion: The operating experience issues that are identified shall be reviewed an evaluated. These include

- Human performance issues, problems and sources of human error shall be identified

- Design elements which support and enhance human performance shall be identified

This evaluation will address in turn each category of issue: USIs, TMI issues, NRC generic letters, AEOD studies, and low-power and shutdown issues.

USIs and TMI issues

Evaluation: The ABB-CE OER treats USIs and TMI issues similarly. They are divided into the following groups of items by ABB-CE: HFE tracking system issues, issues addressed by and incorporated into the System 80+ design, COL applicant issues, and issues that are not applicable (NA) to the System 80+ design. Those classified as going into the HFE tracking system are discussed below in Section 18.4. Those designated as COL applicant issues are listed here:

- Generic issues: GI-57, GI-75, GI-116, GI-117, B-32.
- TMI item: III.A.1.2.
- Generic letters: 91-06 and 91-11.

Further, ABB-CE will include in Chapter 1 of CESSAR-DC a summary list of all COL applicant issues. Those issues designated as NA were reviewed on a sampling basis and no problems were identified.

The next paragraphs discuss those items identified as incorporated into the design. The discussions in the DSER, the OER, and CESSAR-DC were reviewed. In the first version of the OER, the issues appeared to be resolved by hardware/systems types of fixes. Details of just how an item was resolved in the design were somewhat sketchy, especially concerning the HF aspects of the resolution. The references contained only the material which generated the unresolved issue and not the technical findings documents, which resolve or partially resolve the issue. The revised OER provided considerably more detail for the TMI and USI issues and focused on the HF aspects of the issue. The additional details in which the design addresses the particular concerns of the identified USIs and GSIs are acceptable and hence, this criterion is satisfied.

NRC Generic Letters

Evaluation: Three generic letters (GL) are addressed in the OER. For GL 91-06 and 91-11, ABB-CE states that monitoring, surveillance, equipment status, and testing will be COL applicant issues. This appeared to be a very broad transfer of responsibility to the COL. Certainly there are HF aspects of these four areas which need to be addressed in the design. To defer all consideration of these issues to procedural type resolutions that a COL would create does not seem appropriate. As a result of these observations, ABB-CE revised the OER to note those aspects of maintenance and testing that would be addressed in the design. This is further covered in the review of the HFPP and in the review of the ABB-CE "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (Reference 1 of CESSAR-DC Section 18.6) under HFE PRM Element 6. Hence, this criterion is satisfied.

GL 07 on RCP seals is discussed in the section on HFE PRM Element 6, "HSI design."

AEOD Studies

Evaluation: The HFE PRM specifies a review of recent AEOD studies in the human performance area. A brief discussion of this report series was contained in the OER. Additional detail as to how the items identified were incorporated into the design was identified as being desirable. In the revised OER, Appendix A was modified to explain in greater detail the design resolution of the various issues raised in the AEOD series of reports. Hence, this criterion is satisfied.

Low-Power and Shutdown Issues

Evaluation: ABB-CE's review of this area is described in a separate report, System 80+ Shutdown Risk Evaluation Report (Reference 1 of CESSAR-DC Section 18.10, LD-92-076), June 16, 1992. Based upon a brief review, the document was deemed to be reasonably thorough and comprehensive. The list of reference documents was also appropriate and extensive. One item noted was, that considering the OER commitment to continue to review new information and documents, two new documents would be particularly valuable to include. They are: the final version of NUREG-1449 and the December 1991, NUMARC 92-106, "Guidelines for Industry Actions to Assess Shutdown Management." ABB-CE agreed with this comment. Hence, this criterion is satisfied.

18.3.3.2.7 Interview Topics

Criterion: This item lists the topics which should be included in the operator interviews.

Evaluation: The original version of the ABB-CE OER stated that System 80 operator interviews were not conducted. Input from operators was used. The staff was concerned that interviews with System 80 operators were not conducted, since System 80 is the direct predecessor of the System 80+.

As a result of this concern, the reviewers visited operating System 80 plants and conducted interviews with licensed System 80 operators regarding their experience with their plant. BNL Technical Report E2090-T2-5-3/93 was then issued which details the results of these interviews and raises a number of questions and issues relative to the incorporation of System 80 experience into the System 80+. ABB-CE agreed to review those issues. In the first review, ABB-CE noted for each issue whether the issue was already addressed in the design. If the issue was not in the design, then ABB-CE either entered it into the HF issues tracking system or designated it as a COL applicant issue. This information, designating which category each issue fit into, was provided to the staff in draft form and constituted the initial basis for closure of this item, as documented in the minutes of the April 6, 1993, conference call between the staff and ABB-CE. ABB-CE later provided the staff with a report (Ref. 4 of CESSAR-DC Section 18.10, LD-93-135, Attachment 1) that explains in some detail how and why the identified issues are addressed. This report was reviewed by the staff. The report was noted to be comprehensive in analyzing the issues raised by the operator interviews. Many of the issues were not yet fully addressed in the System 80+ design; however, ABB-CE has noted that they either will be addressed by the ongoing design process or they have been specifically added to the HFE tracking system to ensure that they will be addressed later. The combination of activities described above satisfies the PRM criterion.

18.3.3.2.8 Literature Review

Criterion: The review shall include a literature review.

Evaluation: From the documents listed in the OER, it appears that a substantial literature review was conducted. However, the list of references was lacking in documents from ABB-CE System 80 plants. Since System 80 is the direct predecessor to System 80+ it is especially important to consider System 80 experience. As an example, there appears to be valuable information in the System 80 LERs, as noted in Reference 4 of CESSAR-DC Section 18.10, LD-93-135, Attachment 1. As a result of this observation, the reviewers performed a search of System 80 LERs, using the sequence coding and search system, to identify human errors of various sorts and inadequate HSI. Several hundred LERs with such human errors were identified. The recent Licensee Event Reports (1988 to present) were obtained, and reviewed by the staff to ensure a broad range of human performance concerns were represented, and forwarded to ABB-CE for their review. ABB-CE categorized these LERs as follows: design resolution already provided, incorporated into the HF issues tracking system, or not applicable. The results of the review are contained in Appendix B of the revised CE OER. During discussions between the staff and CE during the conference call on April 6. 1993, CE presented examples of how the issues were categorized and either resolved through the design process or identified in the issues tracking system. The staff agreed that the categorization scheme was adequate and that CE would consider these issues as the design process continued. Based on these discussions and the acceptance of the detailed design process for the development of the CE System 80+ main control room, the staff found CE's approach to considering these issues acceptable. Therefore, this issue is resolved.

18.3.3.2.9 Sources

Criterion: This item identifies those industry-wide and plant or subsystem relevant sources that should be included in the OER.

Evaluation: This criterion was satisfied in the DSER, with the following discussion. "Attachment 3, to the applicant's letter of May 8, 1992, satisfactorily identified the various sources used by the applicant to complete its operating experience review. However, the attachment failed to provide the results of the OER that were incorporated into the System 80+ design." Therefore, the relevant sources were considered to be satisfied in the DSER, but the full OER report was developed by ABB-CE to address the second half of the above discussion in the DSER.

18.3.3.2.10 Tracking System

Criterion: Each operating experience issue shall be documented in the HFE tracking system.

Evaluation: Section 2.0 of the OER states that any unresolved design issues identified during the reviews, which may impact the design, will be entered into the HFE tracking system for subsequent resolution and documentation. Section 5.0 of the OER states that the tracking system was implemented in early 1992. Many items in various sections and in the Appendices of the OER are listed as being included in the tracking system. This all appears appropriate and programmatically the tracking system satisfies the PRM criterion.

On a trip to ABB-CE, in May 1993, the tracking system was reviewed by the staff. Selected items from the OER were verified to be included as stated. One item noted was that the information included in the tracking was somewhat sparse and the reference to the original document did not include the section or page. This could make later understanding of the issue difficult. ABB-CE corrected this during the visit, leading to a full satisfaction of the PRM criterion.

18.3.3.2.11 Reference Documents

Criterion: This item lists four documents that the OER program should use in developing the implementation plan.

Evaluation: The OER has satisfactorily used the four identified documents.

18.3.4 Operating Experience Review Findings

An evaluation of the original "ABB-CE Operating Experience Review for System 80+ MMI Design" was completed using the HFE PRM as guidance. Overall, the ABB-CE OER was quite impressive and showed a detailed review of many aspects of pertinent commercial NPP experience, and the subsequent incorporation of appropriate design features into the System 80+ design. Not all aspects of the HFE PRM were completely addressed, however, and so ABB-CE worked with the staff to address the identified concerns. ABB-CE performed additional reviews of areas of operating experience which resulted in items being added to the HF issues tracking system for later incorporation into the design. This additional work is discussed in the revised OER, which also better describes how ABB-CE had already incorporated operating experience into the System 80+ design. The revised ABB-CE OER meets the guidance in the HFE PRM and is acceptable.

18.4 Functional Requirements Analysis and Allocation

18.4.1 Objectives

The HFE PRM for advanced evolutionary reactors specified that a functional requirements analysis (Element 3) and function allocation (Element 4) should be performed. The objective of this review is to provide an evaluation of ABB-CE's functional requirements analysis and function allocation for the System 80+. ABB-CE presents its analyses in "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX80-IC-RR790-02, March 15, 1993).

ABB-CE stated that full analyses of functional requirements and function allocation are not necessary because the System 80+ design is an evolution of the System 80 design that was previously reviewed and approved by the NRC and has an operating history (Palo Verde Units 1, 2, and 3). In addition, ABB-CE stated that both the definition and allocation of functions for the System 80+ are largely unchanged from its predecessor, the System 80. The reviewers agree that the HFE PRM model required tailoring to the System 80+ design; therefore, HFE PRM model modification was also part of this review.

18.4.2 Methodology

The following is a brief chronology of activities that occurred during this review:

(1) Conducted an initial review of ABB-CE's document, "Human Factors Evaluation and Allocation of System 80+ Functions." Reviewers prepared a set of questions to obtain additional information and clarification of several issues. Held a telephone conference call on January 15, 1993, between the staff and ABB-CE to discuss these questions.

(2) Modified the HFE PRM to better tailor the acceptance criteria to the review of an evolutionary design that is a close descendent of an NRC-evaluated design with an operating history. The model revision is briefly described in Section 18.4.3 below and is contained in BNL Technical Report E2090-T1-3-3/93.

(3) The reviewers performed a comparison of the System 80 and System 80+ designs to assess the degree to which the two designs are similar. This was necessary since the revised HFE PRM focuses the review of Elements 3 and 4 largely on function and function allocation differences between a new and predecessor design. The results of this review are presented in Appendix B of BNL Technical Report E2090-T1-3-3/93.

(4) Conducted a preliminary review of ABB-CE's Element 3 and 4 analyses was conducted addressing both the DSER issues and the revised HFE PRM criteria.

(5) Held a telephone conference call on February 16, 1993, between the staff and ABB-CE to discuss the preliminary review and design differences between the System 80 and the System 80+. ABB-CE submitted a February 11, 1993 revision to its draft document entitled "Human Factors Evaluation and Allocation of System 80+ Functions."

(6) Held a telephone conference call on February 19, 1993, between the staff and ABB-CE to discuss the revised ABB-CE document and design differences between the System 80 and System 80+.

(7) ABB-CE submitted a February 23, 1993 revision to its draft document entitled, "Human Factors Evaluation and Allocation of System 80+ Functions."

(8) Conducted a review of the draft document was conducted and presented the results in BNL Technical Report E2090-T2-2-3/93.

(9) Based on this review, ABB-CE issued a final version of its document entitled, "Human Factors Evaluation and Allocation of System 80+ Functions" (dated March 26, 1993).

(10) Reviewed the final version of "Human Factors Evaluation and Allocation of System 80+ Functions." The following are the results of that review.

18.4.3 Revised Human Factors Engineering Program Review Model Elements 3 and 4

The functional requirements analysis described in Element 3 of the HFE PRM is important to the HF review process because it provides a description of the major functions that must be performed to ensure safe operation of a NPP. In Element 4 of the HFE PRM, Allocation of Function, portions of the control function are assigned to the human operators and the plant control system. This assignment should be based on an analysis of the inherent strengths and weaknesses of humans and machines (hardware and software subsystems) as controllers and the synergy that may be achieved through a joint human-machine control system.

The HFE PRM was revised to tailor the criteria to an evolutionary plant in which the functional requirements and allocations of the new design are based largely on the predecessor design. The key modifications to the HFE PRM include

- Greater reliance on evaluating the operating experience of the predecessor plant
- Greater reliance on evaluating functional differences between the System 80+ and the predecessor plant
- A reduced scope of analysis that focuses on (1) differences between the System 80+ and the predecessor plant in terms of functional requirements and function allocation and (2) functions that are unchanged between the System 80 and System 80+ but have had problems identified through an OER

18.4.4 Evaluation of Element 3 - Functional Requirements Analysis

This section provides a review of ABB-CE's definition and analysis of System 80+ functions that are important to plant safety. The allocation of these functions to personnel and plant systems is addressed in Section 18.4.5.

18.4.4.1 ABB-CE Documentation

Two main documents were used as the basis for the review:

- September 10 and 11, 1992, public meeting minutes, dated October 21, 1992, meetings between NRC and ABB-CE.
- Reference 7 of CESSAR-DC Section 18.4, LD-93-056, "System 80+ Human Factors Engineering," Attachment, "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX80-IC-RR790-02, March 15, 1993), ABB-CE letter dated March 26, 1993.

Element 3 is also discussed in the ABB-CE HFPP and CESSAR-DC, although these documents were not the focus of this review.

18.4.4.2 DSER Review

18.4.4.2.1 DSER Issue

In the staff's initial review of this element, it was concluded that ABB-CE had not documented the system functions and identified DSER Issue 18.5.2 - Function Analysis. At the September 10 and 11, 1992, public meeting, ABB-CE agreed to address the DSER issue by preparing a document addressing the following three items

(a) Describing the baseline system (System 80), its functional requirements and the changes and additions to those requirements for the new system (System 80+)

(b) Identifying the system objectives, performance requirements, and constraints of the predecessor system

(c) Identifying any changes to operator performance requirements for the new system

The ABB-CE document, "Human Factors Evaluation and Allocation of System 80+ Functions," was prepared specifically to address the DSER issue. The primary focus of the document is on safety functions (which are described in terms of the ABB-CE critical functions) and success paths.

18.4.4.2.2 DSER Issue Resolution

The DSER issue evaluation focuses on ABB-CE's provision of the three items identified above.

Item (a)

Issue: Describe the baseline system, its functional requirements and the changes and additions to those requirements for the new system.

Evaluation: The baseline system discussed is the System 80. Palo Verde Units 1, 2, and 3 are the only operating NPPs of the System 80 design. The discussion of functional requirements emphasizes safety functions, which are described in terms of CSFs. Non-safety functions are not discussed in much detail. A comparison of functions between the System 80 and the System 80+ are shown in Table 1 of "Human Factors Evaluation and Allocation of System 80+ Functions." Table 2 of that document provides a comparison of safety grade and non-safety grade success paths associated with CSFs of the System 80 and System 80+ plants. Narrative descriptions are provided of the success paths for the System 80+ plant. This criterion is satisfied because these descriptions described the baseline system (System 80+), its functional requirements, and the changes and additions to those requirements for the System 80+.

Item (b)

Issue: Identify the system objectives, performance requirements, and constraints of the predecessor system.

Evaluation: Table 1 of "Human Factors Evaluation and Allocation of System 80+ Functions" lists the CSFs and their purposes (objectives). Table 2 provides a comparison of System 80 and System 80+ in terms of critical functions and their respective success paths. Success paths are components and resource commodities that satisfy particular safety functions. Table 3 shows changes in the evolution of System 80 to System 80+. The following CSF success paths were identified as new:

- Rapid depressurization (RCS heat removal function)
- Hydrogen ignitors (containment environment function)

The following CSF success paths were identified as modified:

- Alternate generator (vital auxiliaries function)
- Startup feed (RCS heat removal)

No CSF success paths were identified as deleted.

The reviewers performed a comparison of the System 80 and System 80+. ABB-CE discussed these differences with the reviewers during telephone conference calls on February 16 and 19, 1993, and explained the relationship of these differences to the CSFs and success paths described in "Human Factors Evaluation and Allocation of System 80+ Functions." Based on this information, ABB-CE has adequately addressed the HFE PRM Element 3 requirements to describe the functional basis of the System 80+. New and modified success paths will receive more detailed analyses during Element 5 Task Analysis to examine the operator's role and the allocation of functions.

With regard to externally dictated performance requirements, Section 2 of "Human Factors Evaluation and Allocation of System 80+ Functions" presents a set of federal regulations, industry standards, and regulatory guidelines from which a set of criteria presented in Section 3.3 were derived. These criteria address control of safety functions via manual and automatic means. These criteria provide useful man-machine function allocation considerations beyond those imposed internally by the design of the plant. Hence, this issue is resolved.

Item (c)

Issue: Identify any changes to operator performance requirements for the new system.

Evaluation: Changes in the operator performance requirements for new and changed success paths are described in Section 4 of "Human Factors Evaluation and Allocation of System 80+ Functions." Hence, this issue is resolved.

18.4.4.3 HFE PRM Criteria-Based Evaluation

The following is a review of the ABB-CE function requirements analysis based upon the revised HFE PRM criteria for Element 3.

HFE PRM Criterion 1.

Criterion: High-level plant safety goals and requirements shall determine the plant safety functions.

Evaluation: Table 1 of ABB-CE's "Human Factors Evaluation and Allocation of System 80+ Functions" provides a listing of high-level safety functions. The purpose statement for each function illustrates a relationship to higher-level safety goals. Hence, this criterion is satisfied.

HFE PRM Criteria 2 and 3.

Criterion 2: Safety-related functions shall be defined (i.e., those functions required to achieve major system performance requirements; or those functions which, if failed, could degrade system or equipment performance or pose a safety hazard to plant personnel or to the general public).

Criterion 3: Safety-related functions of the new plant shall be compared to the predecessor plant to document (1) those that are new, (2) those that have been changed, and (3) those that have been deleted. These shall be referred to as the "modified" functions. Safety-related functions that have not been modified shall be documented as unchanged.

Evaluation: Plant CSFs and their respective success paths are described in "Human Factors Evaluation and Allocation of System 80+ Functions." This criterion is satisfied because Table 3 of the aforementioned document identifies success paths as new, modified, deleted, and unchanged.

HFE PRM Criterion 4.

Criterion: The technical basis for function modification shall be documented. Modified safety functions shall be identified and any functional interrelationships with non-safety systems shall be identified.

Evaluation: This criterion is satisfied because the technical basis for function modification, modified safety functions, and inter-relationships with non-safety systems have been explained.

HFE PRM Criterion 5.

Criterion: Modified safety functions shall be defined as the most general, yet differentiable means whereby the system requirements are met, discharged, or satisfied. Functions shall be arranged in a logical sequence so that any specified operational usage of the system can be traced in an end-to-end path.

Evaluation: Modified safety functions have been identified and described as noted in this criterion. Hence, this criterion is satisfied.

HFE PRM Criterion 6.

Criterion: Modified safety functions shall be described initially in graphic form. Function diagramming shall be done at several levels, starting at "top level" safety goals where a very general picture of major functions is described, and continuing to decompose major functions to several lower levels until a specific critical end-item requirement will emerge, e.g., a piece of equipment, software, or an operator. The functional decomposition should address the following levels:

- High-level safety goals (e.g., maintain RCS integrity)
- Critical safety functions (e.g., maintain RCS pressure control)
- Individual plant systems
- Specific plant components

Evaluation: This criterion describes a function diagramming method that graphically illustrates the hierarchical relationships of plant functions. In "Human Factors Evaluation and Allocation of System 80+ Functions," Figure 2, Goals-Means Relationships, is an example of this type of diagram. Figure 3 illustrates the relationship of the System 80+ success paths to the higher-level goals. Although it does not extend to the level of specific components, it generally satisfies the HFE PRM requirement of a graphic description of safety function. Hence, this criterion is satisfied.

HFE PRM Criterion 7.

Criterion: Detailed narrative descriptions shall be developed for each of the identified modified functions and for the overall system configuration design itself. Each modified function shall be identified and described in terms of inputs (plant condition in which operations are needed), functional processing (control process and performance measures required to achieve the function), outputs (indications of functional operation), feedback (indication of higher-level goal achievement), and interface requirements so that subfunctions are recognized as part of larger functional areas.

Evaluation: The descriptions of the CSFs and success paths are provided in Sections 4.2 and 4.4 of "Human Factors Evaluation and Allocation of System 80+ Functions". This criterion is satisfied because the inputs, functional processing, outputs, and feedback are described for new and modified functions. This detailed information was not provided for those CSFs and success paths of the System 80+ that were considered to be essentially the same in the System 80 because the System 80 plant is an NRC-approved design. Interface requirements were not provided in these descriptions because they were established through ABB-CE's functional task analysis, which was reviewed and found acceptable in Section 18.5. Therefore, this criterion is satisfied.

HFE PRM Criterion 8.

Criterion: Functional operations or activities shall include:

- detecting signals
- measuring information
- comparing one measurement with another
- processing information
- acting upon decisions to produce a desired condition or result on the system or environment (e.g., system and component operation, actuation, and trips)

Evaluation: The functional analysis describes the conditions under which the success paths are required and, in some cases, the parameters that indicate these conditions. Considerations related to detecting, measuring, comparing and processing values are generally not addressed in this document. More detailed analyses are conducted during task analysis, which is addressed by HFE PRM Element 5. Hence, this criterion is satisfied.

HFE PRM Criterion 9.

Criterion: The function analysis shall be kept current over the life cycle of design development.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions" does not define any procedure for keeping the function analysis current. However, Section 6.0 - Conclusions of the ABB-CE document states, "Evaluation of the interaction between the human and machine elements of the plant control system, and the resolution of specific problems identified, will continue as part of Task Analysis, PRA, Verification and Validation, and procedure development activities." Hence, this criterion is satisfied.

HFE PRM Criterion 10.

Criterion: The technical basis upon which the function analysis was performed shall be documented.

Evaluation: ABB-CE has stated that the document, "Human Factors Evaluation and Allocation of System 80+ Function" was written after the function allocation was complete. In Section 5 - Results, ABB-CE states that this document provides a descriptive evaluation that does not aim to create or revise the System 80+ design. The document provides adequate discussions of key considerations that determined the allocation of functions. Hence, this criterion is satisfied.

18.4.4.4 Element 3 Findings

"Human Factors Evaluation and Allocation of System 80+ Functions" describes the critical functions and the success paths that are responsible for satisfying the safety functions. Comparisons are made at a high level between the System 80 and the System 80+ designs and differences are noted.

The following CSF success paths were identified as new:

- Rapid depressurization (RCS heat removal function)
- Hydrogen ignitors (containment environment function)

the following CSF success paths were identified as modified:

- Alternate generator (vital auxiliaries function)
- Startup feed (RCS heat removal)

No CSF success paths were identified as deleted. ABB-CE has stated that these changes should impose little change on the role of the operator compared to the operator's role in the System 80. It was concluded that from a functional basis the System 80 and System 80+ plants were very similar and that the functional differences should not result in major changes in the role of the operator.

18.4.5 Evaluation of Element 4 - Allocation of Function

This section provides a review of how functions were allocated to personnel and plant systems in the System 80+.

18.4.5.1 ABB-CE Documentation

The following ABB-CE document was reviewed:

- Reference 7 of CESSAR-DC Section 18.4, LD-93-056, "System 80+ Human Factors Engineering," Attached, "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX-IC-RR790-02), Rev. 01, March 15, 1993), ABB-CE letter dated March 26, 1993.

The following documents were consulted as a part of this evaluation:

- September 10 and 11, 1992, public meeting minutes, dated October 21, 1992, meetings between NRC and ABB-CE.

- Reference 7 of CESSAR-DC Section 18.4, "System 80+ Human Factors Engineering Issue Closeout," Attached, "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX-IC-RR790-02, Rev. 00, February 23, 1993).

18.4.5.2 DSER Review

18.4.5.2.1 DSER Issues

The staff's initial review of this element concluded that ABB-CE had not confirmed that trade-off studies have been conducted to determine adequate configurations of personnel and system-performed functions. DSER Issue 18.6, "Function Allocation," was identified. At the September 10 and 11, 1992, public meeting, ABB-CE agreed to address the DSER issue in the document, "Human Factors Evaluation and Allocation of System 80+ Functions" by:

(a) describing the baseline system, its function allocation, and the changes and additions to function allocation for the new system,

(b) confirming that trade-off studies or other means have been used to determine adequate configurations of personnel and system performed functions,

(c) confirming that personnel can properly perform tasks allocated to them,

(d) providing auditable details regarding the bases used to allocate functions to the (1) operator, (2) manually-operated controls, (3) automatic control processes, and (4) computer.

18.4.5.2.2 DSER Issue Resolution

The DSER issue evaluation will focus on ABB-CE's provision of the four items identified above as necessary to resolve the issue.

<u>Item (a)</u>

Issue: Describe the baseline system, its function allocation, and the changes and additions to function allocation for the new system.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions" provides a comparison of the System 80+ to a baseline system - the System 80. Table 1 of that document lists the names and purposes of safety functions for the System 80 and System 80+. Table 2 provides a comparison of safety grade and non-safety grade success paths for the System 80 and System 80+. Table 3 describes the success paths of the System 80+ as either unchanged, modified, new or deleted relative to the System 80. Table 4 describes the function allocation of the success paths of the System 80+. This document provides sufficient information to describe the functional basis of the System 80+ and satisfy DSER Issue 18.6, Item a. Hence, this issue is resolved.

<u>Item (b)</u>

Issue: Confirm that trade-off studies or other means have been used to determine adequate configurations of personnel and system performed functions.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions" provides the results of a set of function allocation analyses that evaluated functions performed by personnel and plant systems. ABB-CE stated verbally that these analyses were conducted after function allocations had been completed, as a form of verification. This evaluation examined the trade-offs between allocating functions to personnel and plant systems and determined that the function allocations for the System 80+ were consistent with human factors principles of function allocation. Therefore, this issue is resolved.

<u>Item (c)</u>

Issue: Confirm that personnel can properly perform tasks allocated to them.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions" provides the results of a set of function allocation analyses which justifies the allocation of functions to the operator. Further confirmation that personnel can properly perform tasks allocated to them will be addressed during task analysis and verification and validation. Hence, this issue is resolved.

<u>Item (d)</u>

Issue: Provide auditable details regarding the bases used to allocate functions to the (1) operator, (2) manually-operated controls, (3) automatic control processes, and (4) computer.

Evaluation: Section 4.4 - Allocation Data of Human Factors Evaluation and Allocation of System 80+ Functions provides details related to the allocation of functions to the operator. Important descriptive material includes (1) Table 4, which provides a summary of safety function allocations, and (2) the "Allocation Rationale" discussion that is associated with each success path.

In response to review concerns, ABB-CE provided additional information regarding Table 4 including:

- A description of the five categories of allocation for control functions: automatic, automatic-AND-manual (AAM), manual, manual-OR-automatic, and manual-XOR-automatic (MXA) used in column 7.

- A clarification of the information presented on column 4, which provides the rationale for each function allocation based on a set of function allocation criteria presented in Appendix B of "Human Factors Evaluation and Allocation of System 80+ Functions." This set of criteria were derived from criteria presented in NUREG/CR-3331.

Based on this additional information, the function allocation description was found to be acceptable. Hence, this issue is resolved.

18.4.5.3 HFE PRM Criteria-Based Evaluation

The following is a review of the ABB-CE function requirements analysis based upon the revised HFE PRM criteria for Element 4.

<u>HFE PRM Criterion 1</u>.

Criterion: Functions that were identified as unchanged in Element 3 shall be reviewed to determine (1) those for which the human-machine allocation is unchanged, and (2) those for which the human-machine function allocation has changed (e.g., through the increased use of automation). This latter group shall be described as having "modified" function allocations.

Evaluation: ABB-CE addressed this requirement in Table 3 of Allocation Data of Human Factors Evaluation and Allocation of System 80+ Functions by explicitly categorizing success paths as unchanged, modified, new, and deleted. However, in the terminology developed in the revised version of HFE PRM Elements 3 and 4 the term "modified" refers to those functions that are new, changed, or deleted. ABB-CE's use of the term "modified" corresponds to the term "changed" used in the HFE PRM. This difference is noted because it is a potential source of confusion. Hence, this criterion is satisfied.

<u>HFE PRM Criteria 2, 3, and 5</u>.

Criterion 2: Unchanged functions that have modified function allocations shall be analyzed in terms of resulting human performance requirements based on the expected user population. This analysis should reflect (1) sensitivity, precision, time, and safety requirements, (2) required reliability, and (3) the number and level of skills of personnel required to operate and maintain the system.

Criterion 3: Modified functions (identified in Element 3) shall also be analyzed in terms of resulting human performance requirements based on the expected user population. This analysis should reflect (1) sensitivity, precision, time, and safety requirements; (2) required reliability; and (3) the number and level of skills of personnel required to operate and maintain the system.

Criterion 5: The results of analyses and trade-off studies shall support the adequate configurations of personnel- and system-performed functions. Analyses shall confirm that the personnel element can properly perform tasks allocated to them while maintaining operator situation awareness, workload, and vigilance. Proposed function assignment shall take the maximum advantage of the capabilities of human and machine without imposing unfavorable requirements on either.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions" provides the results of a set of function allocation analyses that evaluate functions performed by personnel and plant systems. This evaluation examined the trade-offs between allocating functions to personnel and plant systems and determined that the function allocations for the System 80+ were consistent with human factors principles of function allocation. Therefore, this criterion is satisfied.

<u>HFE PRM Criterion 4</u>.

Criterion: The allocation criteria, rationale, analyses, and procedures used in the analysis of function allocation shall be documented.

Evaluation: The allocation criteria, rationale, analyses, and procedures used in the analysis of function allocation are presented in "Human Factors Evaluation and Allocation of System 80+ Functions." This generally provides an acceptable description of functions and function allocations. Specific concerns regarding function allocation were adequately addressed in Item b of Section 18.4.5.2.2 of this report. Hence, this criterion is satisfied.

<u>HFE PRM Criterion 6</u>.

Criterion: The OER shall be reviewed to address the case of modified functions. Problematic OER issues shall be considered during the function allocation analyses for modified functions.

Evaluation: The following modified functions were described in "Human Factors Evaluation and Allocation of System 80+ Functions"

- Rapid depressurization (RCS heat removal function)
- Hydrogen ignitors (containment environment function)
- Alternate generator (vital auxiliaries function)
- Startup feed (RCS heat removal)

Section 4.4 of "Human Factors Evaluation and Allocation of System 80+ Functions," indicated that rapid depressurization was provided as an RCS heat removal function in response to problems with PORVs that were identified from previous plant operating histories. The other modified functions were justified on the bases of improved redundancy and diversity or were a return to the allocation of prior plant designs. Hence, this criterion is satisfied.

HFE PRM Criterion 7.

Criterion: The OER shall be reviewed to address the case of unchanged functions that have unchanged function allocations. If problematic OER issues are identified then an analysis shall be performed to (1) justify the original analysis of the function, (2) justify the original human-machine allocation, and (3) identify non-design solutions such as training, personnel selection, and procedure design that will be implemented to address the OER issues.

Evaluation: The analyses performed by ABB-CE did not identify problems with unchanged functions. Problems identified by the OER were generally at the task level (e.g., performing cross-checks of values from redundant sensor channels). These problems were addressed by the HSI conceptual design, which was reviewed by HFE PRM Element 6 (see Section 18.6). Hence, this criterion is satisfied.

HFE PRM Criterion 8.

Criterion: All function allocations shall be reviewed to evaluate the effect of new function allocations on unchanged function allocations.

Evaluation: "Human Factors Evaluation and Allocation of System 80+ Functions," indicates that overall the CSF have not changed between the System 80 and the System 80+. It indicates further that the CSF success paths and their control allocations are similar in System 80 and System 80+ with few changes and additions. The following CSF success paths were identified as new:

- Rapid depressurization (RCS heat removal function)
- Hydrogen ignitors (containment environment function)

The following CSF success paths were identified as modified:

- Alternate generator (vital auxiliaries function)
- Startup feed (RCS heat removal)

Rapid depressurization is used in conjunction with the safety injection system/Direct Vessel Injection to accomplish "once-through-cooling" for beyond-design-basis accidents.

Hydrogen ignitors are used as a backup to the hydrogen purge and recombiners systems.

The alternate generator provides additional redundancy and diversity to the other AC power success paths during loss-of-offsite-AC-power events.

The startup feed provides RCS heat removal during low-power conditions (0 to 5-percent power). It is an alternative to the emergency and main feedwater systems. Hence, this criterion is satisfied.

HFE PRM Criterion 9.

Criterion: Functions shall be re-allocated in an iterative manner, in response to developing design specifics, operating experience, and the outcomes of ongoing analyses and trade studies.

Evaluation: In Section 6.0 - Conclusions of "Human Factors Evaluation and Allocation of System 80+ Functions," ABB-CE states that the evaluation of the interaction between the human and machine elements of the plant control system will continue as part of task analysis, PRA, V&V, and procedure development activities. Hence, this criterion is satisfied.

HFE PRM Criterion 10.

Criterion: The technical basis upon which the function allocation analysis was performed shall be documented.

Evaluation: Appendix B of "Human Factors Evaluation and Allocation of System 80+ Functions" provides an acceptable technical approach to function allocation. The ABB-CE document describes the allocation of functions in terms of this approach. Hence, this criterion is satisfied.

18.4.5.4 Element 4 Findings

The function allocation analyses provided by ABB-CE in "Human Factors Evaluation and Allocation of System 80+ Functions" were performed after much of the function allocation between personnel and plant systems had been completed for the System 80+. The ABB-CE report provided documentation and justification of the function allocation from the System 80+ design process, rather than an analysis of an allocation process that was "in progress." The justifications for function allocations were found to be acceptable. Some specific information requirements of the HFE PRM were not addressed by "Human Factors Evaluation and Allocation of System 80+ Functions" but were adequately addressed by ABB-CE's task analysis methodology. The task analysis methodology is reviewed in FSER Section 18.5.

18.4.6 Summary of Findings For Elements 3 and 4

The ABB-CE report "Human Factors Evaluation and Allocation of System 80+ Functions" addresses the requirements for HFE PRM Element 3 - System Functional Requirements by documenting important functions of the System 80+ and comparing these to the System 80. This review found much functional similarity between the System 80 and System 80+ designs. New success paths included rapid depressurization and hydrogen ignitors, changed success paths included the alternate generator and startup feed.

The ABB-CE report also addressed the requirements of HFE PRM Element 4 through its justification of the allocation of functions between personnel and plant systems. Therefore, Elements 3 and 4 are resolved.

18.5 Task Analysis

The NRC HFE PRM for advanced evolutionary reactors specified that a task analysis (Element 5) should be performed. ABB-CE described their task analysis methodology in Section 18.5 of CESSAR-DC and also in "System 80+ Function and Task Analysis Final Report" (dated January 1989, docketed April 8, 1992). This methodology is referred to in this section as FTA - old method. The staff, in the DSER, identified deficiencies in the scope and depth of analyses provided by ABB-CE. This was identified as DSER Issue 18.7. In response to this DSER issue, ABB-CE submitted their proposed task analysis revision for Section 18.5 of CESSAR-DC (Amendment Q). The revised methodology is referred to in this section as SSARFTA.

18.5.1 Objectives

The objectives of this review were to evaluate the scope of the analyses proposed by ABB-CE and evaluate ABB-CE's task analysis methodology. Evaluation of DSER issues resolution was addressed within each objective.

18.5.2 Methodology

18.5.2.1 Material Reviewed

The following ABB-CE documents were reviewed:

- Functional Task Analysis (FTA), Section 18.5 of CESSAR-DC (through Amendment Q), hereafter referred to as SSAR-DCFTA (revised method).

- Reference 4 of CESSAR-DC Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design," (NPX80-IC-DP790-01, Rev. 02, September 29, 1993).

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18, DSER Open Item Response," ABB-CE letter dated January 18, 1993.

- Reference 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment (untitled), attached response to RAI Nos. 620.27 and 620.28, ABB-CE letter dated May 8, 1992.

- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 2, "Nuplex 80+ Compliance with NUREG-0737 Supplement 1 Requirements," ABB-CE letter dated September 23, 1992.

- Reference 8 of CESSAR-DC Section 18.10, LD-93-100, "System 80+ Information for Issue Closure," Attachment 2, Subattachment 2, "Justification of ABB-CE Positions Requested for Closure of Task Analysis," ABB-CE letter dated June 25, 1993.

18.5.2.2 Review Procedure

The following is a brief chronology of activities that occurred during this review:

(1) Conducted a preliminary review of the ABB-CE document, "System 80+ Function and Task Analysis Final Report" (FTA - old method) during the review of HFE PRM Elements 3 and 4 review.

(2) Conducted a more extensive review of this document, Section 18.5 of CESSAR-DC, and other documents during the Element 5 review using HFE PRM criteria.

(3) ABB-CE provided responses to this review in Reference 8 of CESSAR-DC Section 18.10, LD-93-100, Attachment 2 - Justification of ABB-CE Positions Requested for Closure of Task Analysis.

(4) Upon resolution of all review issues, ABB-CE provided a revised FTA methodology in Section 18.5 of CESSAR-DC, Amendment Q (SSARFTA).

18.5.2.2.1 Review Criteria Documents

The following materials were consulted as part of the evaluation:

- American National Standards Institute, ANSI/ANS 58.8, "Time Response Design Criteria for Safety-Related Operator Action," Santa Monica, California.

- Card, S.K., Moran, T.P. and Newell, A. (1983) The psychology of human-computer interaction, New Jersey: Lawrence Erlbaum Associates, pp. 23-97.

- HFE PRM, forwarded to the Commission in SECY-92-299, dated August 27, 1992.

- Letter from T. Wambach (NRC) to ABB-CE, "Public Meeting September 10 and 11, 1992, Regarding Human Factors Engineering (HFE) Design Issues," (Docket No. 52-002).

- Public meeting minutes from September 10 and 11, 1992, meetings between NRC and ABB-CE.

- Rasmussen, J. (1986). Information processing and human-machine interaction: An approach to cognitive engineering. New York: Elsevier Science (North-Holland).

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 01, December 15, 1992), ABB-CE letter dated December 18, 1992.

- Reference 9 of CESSAR-DC Section 18.5, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "System 80+ Function and Task Analysis Final Report" (January 1989), ABB-CE letter dated May 8, 1992.

- U.S Nuclear Regulatory Commission (1992), "Draft Safety Evaluation Report," (NUREG-1492), Washington, D.C.

18.5.3 Evaluation of Element 5 - Task Analysis

The following review is organized into three major sections

- Section 18.5.3.1 - Task Analysis Scope
- Section 18.5.3.2 - Task Analysis Methodology
- Section 18.5.3.3 - Issues Deferred from Element 4

In the staff's initial review, DSER Issue 18.7 was defined. It provided criteria for defining the behavioral requirements of the tasks that personnel are required to perform to achieve the functions allocated to them. At the September 10 and 11, 1992, public meeting, ABB-CE agreed to update its task analysis methodology and incorporate the following modifications into Section 18.5 of CESSAR-DC:

Item A

Full range of operating modes

- include low-power and abnormal events

- provide justification for not including an event on low-power operations during the task analysis for the RCS panel

Item B

Evaluation of function allocation

- document the resolution of the recommendations for re-allocation of function noted on the following pages in Appendix I of the System 80+ Function and Task Analysis Final Report: pp. I-45, I-5, I-7 through I-12

Item C

Critical tasks

- commit to identify through human reliability analysis for System 80+, critical tasks which impact safety, and to complete a task analysis for any such tasks that are identified

Item D

- address the details of review criteria 3 of the HFE PRM Element 5

Item E

- single failure of DPS, DIAS, or IPSO

- provide justification for not completing a task analysis for operation without the DPS

Items F

Position descriptions

- provide position descriptions for people expected to be in the CR during normal, abnormal, and emergency operations

Item G

Provide justification for not completing a task analysis for the following

- interactions between and among the crew in the CR
- interactions between the crew in the CR and other personnel in the plant
- equipment, documentation, and supplies required to support personnel during normal, abnormal, and emergency operations
- information needed for completing tasks or for reconstructing an event that may not be explicitly identified in the generic procedures
- task analyses for maintenance, inspection, and test activities that take place in the CR
- input to personnel training programs

Item H

Task analysis for 1-, 3-, and 6-person operating crews

- discuss ABB-CE's position regarding this issue

Item I

Provide commitment to address the following issues

- maintenance work order tracking and tag out scheme for CR instruments and equipment identified via CRT and flat panel displays
- an account of how operators will track the status of equipment under test, surveillance, or repair
- impact of tracking scheme/system on normal, abnormal, and emergency operations

Items A, B, E, F, G, H, and I are addressed in Section 18.5.3.1 - Task Analysis Scope and Items C and D are addressed in Section 18.5.3.2 - Task Analysis Methodology of this report.

Unless otherwise indicated all ABB-CE responses to the DSER issues and HFE PRM criteria cited in Sections 18.5.3.1 and 18.5.3.2 are from Reference 8 of CESSAR-DC Section 18.10, LD-93-100, Attachment 2 - Justification of ABB-CE Positions Requested for Closure of Task Analysis.

18.5.3.1 Task Analysis Scope

The purpose of this section is to review the proposed scope of ABB-CE's Function Task Analysis effort against

- requirements identified at the September 10 and 11, 1992, public meeting between NRC and ABB-CE in response to DSER Issue 18.7
- review criteria of HFE PRM Element 5

18.5.3.1.1 DSER Issues Related to Task Analysis Scope

At the September 10 and 11, 1992, public meeting, the staff requested that ABB-CE incorporate the task analysis methodology into Section 18.5 - Functional Task Analysis of CESSAR-DC with specific modifications. This section addresses DSER issues that relate to the scope of the task analysis. In some cases DSER items are closely related to specific HFE PRM criteria. In these cases the relevant HFE PRM criteria are referenced and discussions are deferred to the HFE PRM section.

Item A

Criterion: (See review of HFE PRM Criterion 1 in Section 18.5.3.1.2.)

Item B

Criterion: Document the resolution of the recommendations for re-allocation of functions noted on the following pages in Appendix I of the System 80+ Function and Task Analysis Final Report: pp. I-4, I-5, and I-7 through I-12.

Evaluation: ABB-CE noted that the problems identified in the System 80+ function and task analysis final report were corrected through the design of a revised DPS access scheme. In addition, the specific concerns identified in that evaluation were entered into ABB-CE's HF issues tracking system to ensure follow-up evaluation. The original testing will be repeated for the revised design. This issue is resolved through ABB-CE's inclusion of these issues in ABB-CE's HF issues tracking system and because of the design commitment to perform verification and validation of the operating ensemble to verify that the tracking system items have been adequately dispositioned.

Item E

Criterion: Provide justification for not completing a task analysis for operation without the DPS.

Evaluation: Section 18.5.1.5.1 of CESSAR-DC lists 15 event sequences that will be addressed by the SSARFTA (revised method). Event M - Design Basis Failures of DPS and DIAS indicated that this issue will be addressed by the task analysis. This item is acceptably addressed.

Items F

Criterion: Provide position descriptions for people expected to be in the CR during normal, abnormal, and emergency operations.

Evaluation: Reference 11 of CESSAR-DC, LD-92-065, provides a description of the minimum and maximum number of crew members to occupy the main control room during Start-up, Normal Operations, and Post-trip recovery. The minimum crew size for post-trip recovery is defined as three crew members: one operator at the MCC to control normal success paths, one operator at the safety and auxiliary consoles to control emergency success paths, and one senior reactor operator at the CRS console to monitor critical functions and to direct success path strategies. The maximum crew size for these conditions was defined as six crew members: two operators at the MCC, one operator at the safety console, one operator at the auxiliary console, and two operators at the CRS console. This description was found acceptable because the staffing levels are consistent with those defined in 10CFR50.54 and the description has adequate detail regarding the locations and roles of crew members in the control room to allow interpretation of task analysis results and HSI design requirements.

Item G

Criterion: Provide justification for not completing a task analysis for the following

- interactions between and among the crew in the CR
- interactions between the crew in the CR and other personnel in the plant

Evaluation: HFE PRM Element 5 Criterion 3 is related to this DSER issue. It requires that the detailed task descriptions address communication requirements.

The SSARFTA (revised method) addresses operator tasks sequentially. This approach is more conservative with respect to ABB-CE's primary performance criterion, response time, than modelling operator activities with parallel activities performed by multiple operators. In addition, the SSARFTA (revised method) includes a remarks category for recording task requirements such as communication, crew interaction, and task support that are identified during the task analyses. This modification appears in Section 18.5.1.3.3 of CESSAR-DC. The remarks category was found to be an acceptable mechanism for recording important task requirements related to interactions between crew members and between crew members and the rest of the plant and was therefore found to be acceptable.

Criterion: Provide justification for not completing a task analysis for the following

- equipment, documentation, and supplies required to support personnel during normal, abnormal, and emergency operations
- information needed for completing tasks or for reconstructing an event that may not be explicitly identified in the generic procedures
- task analyses for maintenance, inspection, and test activities that take place in the CR
- input to personnel training programs

Each is described below.

G.1 Equipment, documentation, and supplies required to support personnel during normal, abnormal, and emergency operations.

Evaluation: ABB-CE indicated that equipment, documentation, and supplies and similar task support concerns are addressed through the modification of the FTA - old method to include a remarks category. This category will be used to record task support requirements that are identified during the task analysis for specific events. The remarks category is described in Section 18.5.1.3.3 of CESSAR-DC. This modification, which was incorporated into SSARFTA, adequately addresses the DSER issue.

G.2 Information needed for completing tasks or for reconstructing an event that may not be explicitly identified in the generic procedures.

Evaluation: ABB-CE stated the task analysis [FTA - old method] elaborates the finer details of procedural tasks as a basis to assess the sufficiency of the available procedures. Therefore, task analysis does incorporate information that is not "explicitly identified" in the procedure guidelines.

In addition, the scope of SSARFTA (revised method) event sequences addressed in Sections 18.5.1.5.1 and 18.5.1.5.2 of CESSAR-DC have been expanded beyond those reviewed for the DSER. The current set addresses a range of plant conditions, operator tasks (operation and surveillance), I&C failures, and tasks critical to plant reliability (as evaluated through PRA and HRA analyses). This issue is resolved because the selected event scenarios are considered to encompass plant conditions that are not explicitly identified in generic procedures.

G.3 Task analyses for maintenance, inspection, and test activities that take place in the CR.

Evaluation: ABB-CE stated that the effect of CR maintenance, inspection, and test activities on control room operations has been minimized through the I&C design, including features such as redundant indication capabilities (e.g., DIAS and DPS). Therefore, a single HSI failure does not require an immediate need for repair. Therefore, the focus of FTA (both old and revised methods) is on rule-based plant operations activities. Control room maintenance and repair activities are addressed by V&V to ensure that they are adequately supported by the design. This justification was considered acceptable.

G.4 Input to personnel training programs.

(See HFE PRM Criterion 5 in Section 18.5.3.1.2.)

Item H

Criterion: Discuss ABB-CE's position regarding the performance of task analyses for 1-, 3-, and 6-person operating crews.

Response to this DSER issue also addresses HFE PRM Element 5 Criterion 3, which requires that detailed task descriptions address staffing requirements including the number of personnel, their technical specialty, and specific skills.

Evaluation: ABB-CE defines staffing requirements including number of personnel and their technical specialties in Sections 18.3.2 and 18.6.2.2 of CESSAR-DC. ABB-CE's SSARFTA ensures that the HSI supports the operator's input and output requirements and that individual task elements are within human response capabilities. Coordination of activities between crew members is addressed by verification. Together, task analysis and verification adequately address these DSER and HFE PRM issues.

Item I

Criterion: Provide commitment to address the following issues

- maintenance work order tracking and tag out scheme for CR instruments and equipment identified via CRT and flat panel displays
- an account of how operators will track the status of equipment under test, surveillance, or repair
- impact of tracking scheme/system on normal, abnormal, and emergency operations

These issues are addressed below.

I.1 Maintenance work order tracking and tag out scheme for CR instruments and equipment identified via CRT and flat panel displays.

Evaluation: In Reference 7 of CESSAR-DC Section 18.10, LD-93-005, ABB-CE stated that tasks relating to maintenance work order tracking and tagout are not in the task analysis (both old and revised methods) because these tasks will not be performed in the controlling workspace and have no impact on the control room HSI design. A separate facility to support maintenance work tag-out is provided adjacent to the main control room in the System 80+ design.

This justification was found to be acceptable.

I.2 An account of how operators will track the status of equipment under test, surveillance, or repair.

Evaluation: ABB-CE's stated that equipment status data will be input and maintained by personnel other than the operators. The operators will monitor the status of plant components and success paths through the DPS displays and the success path monitoring capabilities. The HSI characteristics for inputting this data was addressed by the Element 6 review of HSI design methods and general characteristics. This issue was resolved through ABB-CE's agreed to describe in greater detail the HSI for inputting these data and through the commitment in the HSI verification and validation commitments in the SSAR.

I.3 Impact of tracking scheme/system on normal, abnormal, and emergency operations.

Evaluation: ABB-CE's stated that the entry of and maintenance of status information will be performed by personnel other than the operators. ABB-CE also stated that the impact of unavailable components on safety and non-safety success paths will be determined by the DPS success path monitoring algorithms and indicated with alarms. CESSAR-DC Section 18.7.1.8.2 states that unavailable status of plant components will be indicated to the operator through various coding schemes of the HSI, such as the use of cross-hatch over component symbols. ABB-CE's position that this activity will have little impact on CR activities was found acceptable.

18.5.3.1.2 HFE PRM Criteria Related to Task Analysis Scope

HFE PRM Criterion 1

Subcriterion 1a: The scope of the task analysis shall include all operations, maintenance, test, and inspection tasks.

Evaluation: CESSAR-DC Section 18.5.1.5.1 (Amendment Q) lists 15 event sequences for the SSARFTA (revised method), which address a range of operational conditions including technical specification surveillances. Maintenance tasks performed in the CR were determined not to be necessary for task analysis (see Item G.3 in Section 18.5.3.1.1) because such tasks are addressed by verification and validation to ensure they are adequately supported by the design. This justification was considered acceptable.

Subcriterion 1b: The analyses shall be directed to the full range of plant operating modes, including start-up, normal operations, abnormal operations, transient conditions, low-power and shutdown conditions. (Note, response to this concern also addresses DSER Item A from Section 18.5.3.1.1.) Item A requests the following modifications to the task analysis methodology:

Full range of operating modes

- include low-power and abnormal operations events
- provide justification for not including an event on low-power operations during the task analysis for the RCS panel

Evaluation: The list of event sequences in CESSAR-DC Section 18.5.1.5.1 was expanded to its current version in Amendment Q which includes 15 events. These sequences adequately address the range of plant operating conditions requested by this criterion.

ABB-CE provided the following justification for not including an event on low-power operations during the task analysis (FTA - old method) for the RCS panel:

> Several distinctly different (low power) events [currently] receive treatment in the System 80+ TA, but the tasks for which TA data were generated using the RCS panel were limited to those with substantial RCS panel interactions. One such "low power" event is plant startup; this was included in the RCS TA work. . . . . All scenario data are entered as updates to a common TA database. Therefore complete coverage of all panels by all scenarios occurs via successive iterations of the TA performed for the remaining panel designs. Impact of information gained through successive iterations will be factored into the RCS panel design.

This criterion is satisfied because low power operations are adequately addressed by the expanded range of event sequences presented in CESSAR-DC Section 18.5.1.5.1.

Subcriterion 1c: The analyses shall include tasks performed in the CR as well as outside of the CR.

Evaluation: The HFPP provides a commitment to perform task analyses for tasks addressed by EOPs that are performed at the remote shutdown panel and local control stations. This commitment satisfies this criterion.

HFE PRM Criterion 5.

Criterion: The task analysis results shall provide input to the personnel training programs.

Evaluation: Item G.4 - Input to Personnel Training Programs from Section 18.5.3.1.1 also addresses training. It requests ABB-CE to provide a justification for not performing task analyses to provide input to personnel training programs. In response to Item G.4, ABB-CE provided the following justification:

> Although the present TA [task analysis methodology] will be a useful input to the COL applicant training program (and will be so provided by ABB-CE), the purpose of the TA is to serve as a design tool. Therefore, it remains a discretionary COL applicant issue as to how the TA database would be best enhanced to support training. This may depend on other aspects of the COL applicant's training program. It is in any case out of ABB-CE's scope for design certification.

(This response applies to both the FTA - old method and SSARFTA (revised method.) ABB-CE provided a commitment in its HFPP to provide vendor task analysis results to the COL applicant. Based on these responses both Criterion 5 and Item G.4 were found to be adequately addressed.

18.5.3.2 Task Analysis Methodology

The purpose of this section is to review the technical basis and typical outputs of the FTA methodology against:

- Requirements identified at the September 10 and 11, 1992, public meeting between NRC and ABB-CE in response to DSER Issue 18.7.
- Review criteria of HFE PRM Element 5.

The results are presented in three sections:

- Section 18.5.3.2.1 - General Comments Related to Task Analysis Methodology
- Section 18.5.3.2.2 - DSER Issues Related to the Task Analysis Methodology
- Section 18.5.3.2.3 - HFE PRM Criteria Related to the Task Analysis Methodology

18.5.3.2.1 General Comments Related to Task Analysis Methodology

Section 4.2 of "System 80+ Function and Task Analysis Final Report" states that the FTA was based in part on the human processor model for simple decision processes (Card et al., 1983). The following concern addresses the application of this model by ABB-CE to the FTA methodology.

> The model human processor uses three estimates of human performance (1) slowman (worst performance), (2) middleman (nominal performance), and (3) fastman (best performance). The discussion provided by Card et al. (pp. 44 - 45) indicates that both the middleman value and the range (fastman-slowman) should be considered when describing human behavior. The criterion used by ABB-CE is based only on the middleman value. This may result in a failure to identify tasks that cannot be performed by operators who have reaction times in the slowman range. ABB-CE was requested to clarify the acceptability and limitations of using the middleman value as a criterion for initial screening of the acceptability of tasks.

Evaluation: ABB-CE's workload criteria, which compare estimates of time available for task elements to estimates of the time required by task elements, have been revised. The former analysis process had a two-level screening process for task elements. The initial screening level was based on the human processor model for simple decision processes and screened task elements using the middleman criterion. Task elements that did not satisfy this criterion were further analyzed using a second screening procedure, which used an unspecified process to examine task elements in greater detail.

The refined analysis process, which appears in Section 18.5.1.4 of CESSAR-DC, also has a two-level screening process. The initial screening level now uses a screening criterion of one minute for each required manual manipulation (task element). This value is based on ANS 58.8, "Time Response Design Criteria for Safety-Related Operator Action." The second screening level is now based on the human processor model and will use explicitly stated conservative assumptions for human and equipment response time performance. During the review it was determined that the middleman criteria of the model was not conservative because it was not representative of those members of the user population who had response times in the slow range. ABB-CE agreed to use the slowman criterion rather than the middleman criterion as a conservative estimate of slow response time. This commitment satisfies this concern because it provides conservative estimates of operator performance.

18.5.3.2.2 DSER Issues Related to Task Analysis Methodology

At the September 10 and 11, 1992, public meeting, the staff requested that ABB-CE incorporate the task analysis methodology into Section 18.5 - Functional Task Analysis of CESSAR-DC with specific modifications. This section addresses DSER issues related to the task analysis methodology. In some cases, DSER items are closely related to specific HFE PRM criteria. In these cases, the relevant HFE PRM criteria are referenced and discussions are deferred to the HFE PRM section.

<u>Item C</u> (See HFE PRM Criterion 2.)

<u>Item D</u> (See HFE PRM Criterion 3.)

18.5.3.2.3 HFE PRM Criteria Related to Task Analysis Methodology

<u>HFE PRM Criterion 2</u>

Criterion: The analysis shall link the identified and described tasks in operational sequence diagrams. A review of the descriptions and operational sequence diagrams shall identify which tasks can be considered "critical" in terms of importance for function achievement, potential for human error, and impact of task failure. Human actions which are found to affect plant risk in PRA sensitivity analyses shall also be considered "critical." Where critical functions are automated, the analyses shall consider all human tasks including monitoring of an automated safety system and back-up actions if it fails.

(Satisfaction of this criterion also satisfies Item C - Critical Task Analysis from Section 18.5.3.2.2. Item C states, "Commit to identify through Human Reliability Analysis for System 80+, critical tasks which impact safety, and to complete a task analysis for any such tasks that are identified.")

Evaluation: The FTA methods (both SSARFTA and FTA - old method) present tasks in timeline diagrams similar to operational sequence diagrams described in the criterion above. This representation satisfies the HFE PRM requirement for operational sequence diagrams.

The FTA methods (both SSARFTA and FTA - old method) identify tasks for which the time required for execution may be in conflict with the time available for execution. However, the FTA methods (both) do not specifically identify tasks that can be considered "critical" in terms of function achievement, potential for human error, and impact of task failure as described in the criterion above. Task elements that do not satisfy the FTA criterion for response time are identified for assessment and resolution through the design process and HFPP.

ABB-CE has recently completed a PRA sensitivity analysis to identify human actions that may be found to affect plant risk. Section 18.5.1.5.2 states that the critical tasks identified through this analysis will be evaluated further using task analysis. The critical tasks are not listed in Section 18.5.1.5.2 of CESSAR-DC because they represent results rather than methodology and are subject to revision with the PRA.

As a result of this PRA analysis, critical operator tasks were identified. DSER Item C states that these critical tasks should be further evaluated using task analysis. ABB-CE commits to do this in Section 18.5.1.5.2 of CESSAR-DC, using SSARFTA. Therefore, ABB-CE has used PRA to identify critical tasks. This approach is acceptable.

<u>HFE PRM Criterion 3</u>

Criterion: Task analyses shall begin on a gross level and involve the development of detailed narrative descriptions of what personnel must do. Task analyses shall define the nature of the input, process, and output required by and of personnel. This criterion is composed of nine subcriteria (a to i). The FTA provides a valuable analysis of information requirements at the gross level. However, additional information was required for the following subcriteria: a, b, c, e, f, g, and h, discussed below.

(Response to this criterion also satisfies Item D - Address details of Criterion 3 in the HFE PRM from Section 18.5.3.2.2.)

Subcriterion 3a: Information Requirements

- information required, including cues for task initiation
- information available

Evaluation: The SSARFTA will identify information requirements for 15 selected events (CESSAR-DC Section 18.5.1.5.1). Information requirements will be derived for individual steps of these events. The description of the individual steps of these events are based on "Combustion Engineering Emergency Procedure Guidelines" (1987). Time lines for these events is based on process time estimates derived by "evaluating data from specific event profiles, based on operator experience and process transient response models" (Section 18.5.1.5.3 of Ref. 2). This criterion is satisfied because the information requirements derived from SSARFTA, described in Section 18.5.1 of CESSAR-DC, address information required and information available.

Subcriterion 3b: Decision-Making Requirements

- description of the decisions to be made (relative, absolute, and probabilistic)
- evaluations to be performed
- decisions that are probable based on the evaluation (opportunities for cognitive errors, such as capture errors will be identified and carefully analyzed)

Evaluation: The FTA models all operator decisions as simple, rule-based behavior. ABB-CE was requested in TA issue 4.2 to describe how complex decision-making, operator errors, and knowledge-based behavior are addressed by their task analysis methodology. Each is described below.

<u>Complex decision-making</u>. ABB-CE stated that the TA deals with decision making as it is structured by the procedures and the operating sequences... Although information requirements necessary to evaluate all procedural decisions (including contingencies) will be addressed, the event sequences and time response evaluations will not necessarily exercise each possible decision contingency. However, care will be taken in the development of TA scenarios to ensure that they address a range of complexity in terms of demands on operator performance, and are not limited to straight-forward or low-demand cases.

This response was found to be acceptable because it provided a commitment to use TA scenarios that vary in complexity with respect to demands on operator performance and therefore ensures that challenging decision-making activities are addressed.

Operator errors. ABB-CE stated that operator errors are addressed by three mechanisms of SSARFTA. First, sections of event scenarios that are considered to have high workload, based on their failure to pass the first screening criteria, are considered error-likely situations and receive a more detailed assessment through the more detailed analysis and screening criteria of the SSARFTA. Second, critical tasks identified through the PRA will be subject to analysis via SSARFTA. Third, observations regarding unique task requirements such as communication, crew coordination, and task support requirements will be recorded using the SSARFTA remarks category.

This was found to be acceptable because situations in which errors are likely are specifically analyzed.

Knowledge-based behavior. ABB-CE stated that the present TA methodology is focused on rule-based (e.g., procedural) rather than knowledge-based (e.g., reasoning) behavior. This reflects the purpose of the TA, which is to support design. In the tasks addressed by the present TA, the plant designer wishes to minimize the need for operators to engage in complex knowledge-based behavior.

While the examination of knowledge-based behavior (reasoning using detailed knowledge of the plant) can be a valuable design tool for ensuring that the HSI supports complex scenarios, such as diagnosis of multiple failures, ABB-CE's focus on rule-based behavior was considered acceptable for identifying basic control and display requirements. Events involving a variety of equipment failures will be addressed during validation. This response was found to be acceptable because the task analysis process addresses basic control and display requirements and because knowledge-based behavior is adequately addressed later in the design process during validation.

Subcriterion 3c: Response Requirements

- action to be taken
- overlap of task requirements (serial versus parallel task elements)
- frequency
- speed/time line requirements
- tolerance/accuracy
- operational limits of personnel performance
- operational limits of machine and software
- body movements required by action taken

The applicability of the FTA to the specific response requirements of Subcriterion 3c are in 3C-1 through 3C-4.

3c-1. The FTA methodology addresses the following response requirements

- actions to be taken
- frequency
- speed/time line requirements
- tolerance/accuracy
- operational limits of personnel performance (comparison of the time available to perform actions to the time theoretically required to perform these actions).

Evaluation: This criterion is satisfied because the above categories of response requirements are specifically defined and analyzed by ABB-CE's CESSAR-DC FTA.

3c-2. The CESSAR-DC FTA (revised method) does not address overlap of task elements. In Section 18.5.1.1 of the revised method ABB-CE states: "The FTA will consider task elements to be additive and serially processed, unless otherwise noted." No general consideration is given to complex interactions of steps or personnel in the FTA.

Evaluation: ABB-CE's response reiterated that both FTA report and CESSAR-DC FTA consider task elements to be additive and serially processed with no complex interactions. This approach is conservative with respect to task completion time - the main performance measure of the methodology. Task interactions will be observed during validation. This response was found to be acceptable.

3c-3. The operational limits of machine and software, such as computer response time, are not addressed by the ABB-CE task analysis.

Evaluation: ABB-CE stated that machine response time is not limiting in the proposed screening model for SSARFTA. The maximum system response time of the HSI (e.g., to call up a particular screen) is two seconds; the task analysis screening model assumes a task completion time of one minute. ABB-CE stated that machine response time may be significant in (and incorporated by) the more detailed analysis which uses a human processor model. ABB-CE stated in Section 18.5.1.4 that these detailed analyses will use explicitly stated conservative assumptions for human and equipment response time performance. This criterion is satisfied because machine response time is accounted for in the SSARFTA at both the task screening level and the detailed analysis level. At the task screening level, machine response time is accounted for by the conservative estimate of task completion time. At the detailed analysis level, machine response time is addressed by ABB-CE's commitment to use conservative estimates of machine response time.

3c-4. The body movements required for operation of the HSI are not adequately addressed since the task analysis does not consider the physical design of the control panels. Only very simple assumptions regarding target size and hand movement distance are made. It does not take into account the effect of the position and frequency of required motions upon operator performance and fatigue.

Evaluation: The analysis of body movements with regard to usability and comfort are addressed through suitability verification and validation. The analysis of body movements with regard to task completion time is addressed during the SSARFTA analysis by the loss of DPS event sequence. This is a limiting scenario because it requires the greatest amount of travel around the controlling workspace to access plant data. The concern for body movements required to operate the HSI is adequately addressed by the use of the loss of DPS event sequence and by the evaluations planned for verification and validation.

Subcriterion 3d: Feedback Requirements

- feedback required to indicate adequacy of actions taken

Evaluation: Feedback requirements are defined by the EPGs for the scenarios addressed by the task analysis. Therefore the issue of feedback is acceptably resolved for the tasks and HSI components addressed by this task analysis.

Subcriterion 3e: Workload

- cognitive
- physical
- estimation of difficulty level

Evaluation: Section 18.5.1.1 of CESSAR-DC states that regarding workload, the main concern in the FTA is with mental tasks in control center activities. The associated physical tasks are within the capabilities of the 5th percentile female operator. Exceptions to this assumption, such as might occur for a locally performed task, are documented in the data.

This criterion is satisfied with respect to mental workload because mental workload is acceptably addressed by the SSARFTA. This criterion is also satisfied with respect to physical workload because the 5th percentile female operator is a conservative criteria for modelling operator characteristics.

Subcriterion 3f: Support Requirements

- special/protective clothing
- job aids or reference materials required
- tools and equipment required
- computer-processing support aids

ABB-CE was requested to describe the degree to which support requirements will be addressed by the task analysis.

Evaluation: Some support requirements have received formal treatment during the review such as procedure storage in RAI Response 620.28 and the adequacy of document lay-down space in HF issue tracking system (TOI) Item 92. Other support requirements will be recorded as remarks category during conduct of the task analysis. Section 18.5.1.3.3 states:

> Remarks accommodate extra notations or miscellaneous task requirements from data categories with infrequent significance. In the present task analysis [SSARFTA], these issues could include, for example, specific workplace suitability issues, task support requirements, communications requirements, crew interactions, or hazard identification.

This criterion is satisfied because these commitments by ABB-CE to evaluate support requirements address an adequate range of support requirements in an acceptable manner.

Subcriterion 3g: Workplace Factors

- workspace envelope required by action taken
- workspace conditions
- location and condition of the work environment

Evaluation: The frequency and distance of movements required of operators in the CR will be addressed by link analysis in which these movements will be recorded and analyzed. Section 18.5.1.5.5 of CESSAR-DC (Amendment Q) states that link analysis will be performed for design basis normal operations and plant shutdown during a loss of the DPS. Loss of the DPS is considered a limiting case because it requires the operator to travel to individual control panels to access data from the DIAS displays.

In addition, workplace factors identified during task analysis that may have important effects on operator performance will be recorded in the remarks category.

Workplace factors are also addressed during the design of HSI systems and components by the requirement to conform to the HFESGB. Conformance will be evaluated during design reviews and suitability verification.

Subcriterion 3h: Staffing and Communication Requirements

- number of personnel, their technical specialty, and specific skills
- communications required, including type
- personnel interaction when more than one person is involved

Evaluation: Unique communication and interaction requirements identified during the SSARFTA analysis will be recorded in the remarks category. The effects of staffing and communication will be evaluated during validation.

Subcriterion 3i: Hazard Identification

- identification of hazards involved

Evaluation: Section 18.5.1.1 of CESSAR-DC states the following task analysis assumption regarding environmental hazards:

> The workspace environments in the main control room and/or remote shutdown room remain habitable for all design basis events and scenarios. However, local control stations included in the FTA shall be individually evaluated for personnel hazards as part of the evaluation of the specified operating sequences and tasks.

The analyses of operator tasks using the detailed processing model, per Section 18.5.1.4 of CESSAR-DC, will use conservative assumptions for human and equipment response time. Hazards will be recorded using the remarks category.

<u>HFE PRM Criterion 4</u>.

Criterion: The task analysis shall be iterative and become progressively more detailed over the design cycle. The task analysis shall be detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment.

Evaluation: Although the SSARFTA was not considered to be highly iterative by nature, it was found acceptable because the results of this process enables specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment in terms of device type, measurement units, and value range, accuracy, and precision.

18.5.3.3 Issues Deferred from Element 4

The following is a review of HFE PRM criteria from Element 4 that were deferred until Element 5 - Task Analysis.

<u>HFE PRM Element 4: Criteria 2, 3, & 5</u>.

Unchanged functions (Criterion 2) and modified functions (Criterion 3) shall be analyzed in terms of resulting human performance requirements. The results of analyses and trade-off studies shall support the adequate configurations of personnel- and system-performed functions (Criterion 5). ABB-CE was requested to describe how these issues will be addressed.

Evaluation: The SSARFTA addresses unchanged and modified functions. ABB-CE stated (TA review issue A.1) that the necessary uses of new and modified functions (i.e., rapid depressurization, hydrogen ignitors, alternate generator, startup feedwater system) are specified in the procedure guidelines and operating sequences employed in the task analysis. ABB-CE stated further in LD-93-100, System 80+ Information for Issue Closure, Attachment 2:

> The analytic scope of the TA [SSARFTA] will exercise the new and modified functions, extend the specified details of the operators' role from the function to the task level, identify human task performance requirements, and assess the resulting task loadings. Excessive loadings will result in further evaluation and formal resolution of the resulting allocation and design issues.

These criteria are satisfied because unchanged functions and modified functions will be analyzed in terms of resulting human performance requirements and functions that pose excessive loadings will be subject to additional analyses to ensure that the configurations of personnel- and system-performed functions are adequately supported.

18.5.4 Task Analysis Findings

This review examined the adequacy of the scope and methodology of ABB-CE's proposed task analysis effort to ensure that it established control and display requirements and evaluated performance requirements imposed on operators.

The task analysis was found to adequately address control and display requirements. The proposed scope of the SSARFTA effort was found to address an adequate range of system failures and plant operating conditions and included critical tasks identified through PRA. Acceptable categories of control and display requirements are generated for individual HSI components including device type, range, accuracy, precision, and units of measure. In addition, the task analysis includes provisions for recording special support characteristics necessary for facilitating operator tasks.

The task analysis was also found to be adequate for addressing performance requirements imposed on operators. The task analysis evaluated task loading by determining whether the time required for task element completion is consistent with the time available for completion. The criteria for estimating the human performance time were revised by ABB-CE as a result of this review. The resulting criteria were found to be acceptable. While this methodology was adequate for addressing operator performance at the individual task element level, it was found that some DSER issues and HFE PRM criteria regarding communication and coordination between multiple crew members were not addressed. However, these concerns are adequately addressed by ABB-CE's verification and validation program. As a result, the proposed task analysis effort was found to be acceptable.

18.6 Human System Interface Design

The objective of HFE PRM Element 6 - HSI Design is to ensure that HFE principles and criteria have been applied along with other design requirements to identify, select, and design the particular equipment to be operated, maintained, and controlled by plant personnel. Element 6 is concerned with design methods, criteria used for making design decisions, interim products (e.g., standard design features) and the final design. HFE PRM Element 8 - V&V will provide a detailed review of the final design.

Review issues related to Element 6 are addressed in three major subsections, each pertaining to separate phases of the HFE PRM Element 6 review:

- Standard Design Features
- Human System Interface Design Methods and General Characteristics
- Human Factors Engineering Standards, Guidelines, and Bases

The first subsection, 18.6.1, "Standard Design Features," provides a review of important elements of the System 80+ design, including six standard design features and the IPSO. The objective of the review was to determine the acceptability of the basic design features of the System 80+ advanced CR as described in the CESSAR-DC and other design basis documents. Further, CR design was reviewed against the Supplement 1 to NUREG-0737 requirements for an SPDS.

The second subsection, 18.6.2, "Human System Interaction Design Methods and General Characteristics," addresses:

- the methods for implementing the display and control requirements, selecting hardware and software, and refining design concepts

- criteria used to determine CR and control panel arrangements including the overall configuration of the main control console and the position of individual control/display devices within individual panels

- general design characteristics that were incorporated into the HSI

The application of the methods and criteria to the design of the CR configuration, RCS panel, and remote shutdown panel is discussed. Relevant DSER issues are also addressed.

The third subsection, 18.6.3, "Human Factors Engineering Standards, Guidelines, and Bases," provides a review of ABB-CE's HFE design criteria. The ABB-CE document primarily addressed by this review is "Human Factors Engineering Standards, Guidelines, and Bases for Nuplex 80+" (NPX80-IC-DR-791-02). This review addressed issues related to the design guidelines for technical basis and validity, guideline integration, and procedure for implementation.

DSER Review

All of the DSER issues that pertain to HFE PRM Element 6 are subsumed in a single issue: DSER Issue 18.8 Element 6 - Human/system Interface Design. This issue contains the following sub-issues:

- 18.8.1 Information Coding Methods Used in the System 80+ Control Room

  - 18.8.1.1 - Shape Coding Used to Prioritize Alarms

  - 18.8.1.3 - Flash Coding of Alarms

  - 18.8.1.4 - Size Coding of Alarms

- 18.8.1.5 - Quality and Types of Information Encoded in the Control Room

- 18.8.2 - Additional HSI Information Required for Staff Review

DSER Issue 18.8.2 contains the following sub-items:

(a) Provide human engineering justification for:

(1) control panel profiles

(2) control panel arrangement in the control room

(3) the selection of control devices

(4) the selection of the display devices

(5) the alarm scheme

(6) the interactive display hierarchy

(7) the number of colors, shapes, and patterns used to convey information in the control room

(b) Provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for:

(1) IPSO

(2) alarm scheme and alarm acknowledgement

(3) readability of alarm text and tiles from all operator positions in control room

(4) display hierarchy and navigation scheme used for CRTs

(5) number of colors and shades used on displays

(6) types and amount of information encoded in the control room as well as the encoding techniques used

(7) audible and tactile feedback for controls, controllers, and other devices

(8) auditable documentation of the design process that supports the human performance aspects of the reduction in the quantity of data presented to the operator

(9) impact of human performance of the difference between breadth of information in System 80 and System 80+ control rooms

(10) qualitative and quantitative criteria that identify when the operator is receiving "enough" rather than "too many" or "too few" number of alarms and displays

(11) auditable documentation to track the data/information that was lost/gained between System 80 and System 80+ control room designs

(12) effects (positive and negative) on operators performance of the changes, individually and collectively, between System 80 and System 80+

The sub-issues and sub-items of Issue 18.8 were individually evaluated and resolved during relevant phases of the Element 6 review. Table 18.4 provides a cross-reference between the DSER sub-issues/sub-items and the appropriate FSER sections.

Table 18.4 Resolution of HSI design DSER issue items

DSER ITEM	FSER SECTION
18.8	18.6.3.3.1.1
18.8.1	18.6.1.3.1.3.1
18.8.1.3	18.6.1.3.1.3.2
18.8.1.4	18.6.1.3.1.3.3
18.8.1.5	18.6.3.3.1.2
18.8.2	
a.1	18.6.2.3.2.2
a.2	18.6.2.3.2.4
a.3	18.6.2.3.2.4
a.4	18.6.2.3.2.4
a.5	18.6.1.3.1.3.4
a.6	18.6.1.3.1.3.4
a.7	18.6.3.3.1.3
b.1	18.6.1.3.1.3.5
b.2	18.6.1.3.1.3.5
b.3	18.6.3.3.1.4
b.4	18.6.1.3.1.3.5
b.5	18.6.3.3.1.4
b.6	18.6.3.3.1.4
b.7	18.6.3.3.1.4
b.8	18.6.2.3.2.1
b.9	18.6.2.3.2.1
b.10	18.6.2.3.2.1
b.11	18.6.2.3.2.1
b.12	18.6.2.3.2.1

18.6.1 Standard Design Features

The following is a review of important elements of the System 80+ design including six standard design features and the IPSO. A complete list of characteristics for each standard feature is provided in CESSAR-DC Section 18.7.1. The review includes DSER issues related to the HSI and CR design compared to the staff's criteria for an SPDS.

18.6.1.1 Objectives

The objective of this review was to determine the acceptability of the basic design features of the System 80+ advanced CR on the basis of their consistency with established human factors standards, guidelines, and principles. The focus was on the acceptability of these features as design elements, as described in the CESSAR-DC and other design basis documents and as represented in the mockups of the MCC and IPSO.

18.6.1.2 Methodology

18.6.1.2.1 Description of Review Methodology

In conducting the design features review the following methodology was used.

First, reviewed relevant design documentation. This included documentation from Elements 1 to 5 of the HFE PRM that addressed the HFPP, OER, function analysis and allocation, and task analysis. In addition, specific reports noted below, that describe the design features, were reviewed.

Second, conducted an onsite review. ABB-CE demonstrated the design features using a mockup and discussed how the RCS panels specifically were developed. The RCS panels provided an example implementation of the design features under review.

Third, selected scenarios for use in evaluating the RCS panel from a functional standpoint. Used these scenarios for panel walkthroughs to determine if necessary controls and displays were available and if the operator could easily access them. Identified the CSF success paths pertinent to the RCS panel for use, namely:

- RCS pressure control
  - Pressurizer heaters & spray
  - Pressure relief
- Core heat removal
  - Natural circulation
  - Forced circulation

Selected two of the EOGs that contained steps relevant to the RCS panel

- Reactor trip
- Loss-of-offsite power

Fourth, Used the above scenarios for a thorough set of walkthroughs at the panels. then the design features were evaluated against available HF standards, guidelines, and principles. In addition, specific displays were selected for more in-depth review using HF guidelines. Performed this review both onsite using the dynamic mockup and in a desktop fashion using the design drawings and descriptions.

Fifth, reviewed the HSI design against selected design concerns that had been identified for System 80 plants via the OER to determine if operating experience was appropriately considered in the design of the System 80+ CR.

Finally, reviewed the HSI design against SPDS requirements identified in Supplement 1 to NUREG-0737.

18.6.1.2.2 Material Reviewed

Used the following ABB-CE documents as the basis for the review:

- Reference 10 of CESSAR-DC Section 18.10, ALWR-92-203, "Review of Human Factors for System 80+ and DCRDR Audit," ABB-CE letter dated April 30, 1992.

- CESSAR-DC Section 18.7.

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18, DSER Open Item Response," ABB-CE letter dated January 18, 1993.

- Reference 9 of CESSAR-DC Section 18.10, LD-93-106, "Nuplex 80+ Design Features Review Comment Responses," Attachment 1, "Design Features Review Comment Responses," ABB-CE letter dated June 30, 1993.

- Reference 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 1, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992.

- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation

Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX-IC-DP-790-01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.

- Reference 1 of CESSAR-DC Section 18.10, LD-92-076, "System 80+ Shutdown Risk Report, Revision 1," attached "System 80+ Shutdown Risk Evaluation Report" (DCTR 10, Draft, June 15, 1992), ABB-CE letter dated June 16, 1992.

18.6.1.2.3 Design Criteria Documents

The following materials were consulted as part of this evaluation:

- Gertman, D., et al., Integrated Process Status Overview (IPSO): Status Report, OECD Halden Reactor Project, HWR-158, April 1986.
- NRC Internal Memorandum, Closure of Issues from the Draft Safety Evaluation Report (DSER) for System 80+, Docket No. 52-002, Letter from B.A. Boger (NRC) to D.M. Crutchfield (NRC), June 14, 1993.
- NRC Internal Memorandum, HICB Review of System 80+ Design Features Related to Instrumentation and Control, M83133, Memorandum from J.S. Wermiel (NRC) to W. Swenson (NRC), June 23, 1993.
- Reference 1 of CESSAR-DC Section 18.6, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).
- Reiersen, C. et al., Further Evaluation Exercises with the Integrated Process Status Overview - IPSO, OECD Halden Reactor Project, HWR-184, April 1987.
- U.S. Nuclear Regulatory Commission (1988), "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," (NUREG-1342), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1991), "Compilation of Alarm System Guidelines and Evaluation of Their Applicability to Hybrid and Advanced Control Rooms," (Draft NUREG/CR-6105), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1992), "Advanced Human/system Interface Design Review Guideline," (Draft NUREG/CR-5908), Washington, D.C.
- U.S. Nuclear Regulatory Commission, "Reactor Coolant Pump Seal Related Instrumentation and Operator Response," (NUREG/CR-4544), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1980, 1982), "Clarification of TMI Action Plan Requirements," (NUREG-0737 and Supplement 1), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1981), "Guidelines for Control Room Design Reviews" (NUREG-0700), Washington, D.C.

18.6.1.2.4 Scope and Limitations

The scope of this review encompassed the following design features:

- DPS display hierarchy
- DIAS alarm tile display
- DIAS dedicated parameter display
- DIAS multiple parameter display
- CCS process controller display
- CCS switch configuration
- IPSO

The first six features are standard in the sense that their basic design will be applied to various panels in the CR. Associated with each standard feature was a set of design characteristics, which were described in Reference 10 of CESSAR-DC Section 18.10, ALWR-92-203. In addition, the IPSO was included in this review. The main control room configuration (MCRC) was not evaluated because the design of the individual panels that comprise it was incomplete.

This review focused on the design basis of the design features and their associated design characteristics. In addition, a limited review of design implementation details was conducted for selected parts of the RCS panel and the chemical and volume control (CVCS) panel.

The ABB-CE mockup of the CR was used in this review. This mockup consisted of selected panels of the MCC in a static representation as well as portions of the RCS and CVCS panels in a dynamic stimulated HSI mockup. This mockup was not driven by a plant simulation. In addition, a static representation of the IPSO was presented via a rear projection display device.

The RCS panel is divided into three functional groups: RCPs on the left, the RCS on the right, and the reactor coolant seal and bleed system in the center. Only the functional group for the center portion of the panel, the RCS portion, was mocked up. Regarding the design features themselves, the DIAS multiple parameter display, the CCS process controller display, and the CCS component controls were not functional on the RCS panel and were observed and operated on the CVCS panel. Additionally, a limited set of the CRT screens of the DPS were designed and not all features of the DIAS displays were fully operational on the mockup. Further, the portions of the design that were completed, have been reviewed by ABB-CE, but findings not yet implemented on the mockup were due to cost, time, and higher priorities. Therefore, discrepancies with ABB-CE's, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (HFESGB, Reference 1 of CESSAR-DC Section 18.6), guidelines were still present in the mockup.

Additionally, the HSI features, as designed and mocked up, were based on the original reactor systems design that was completed several years ago. The systems design has continued to evolve and the HSI mockup did not completely match the latest plant design.

While a final design was not available for the following evaluation, the documentation and mockup permitted an evaluation of the design features at their current level of description. A complete verification and validation of the final CR design is required before the issuance of a COL for a System 80+ plant. This will be addressed by HFE PRM Element 8 - V&V after the design is complete.

18.6.1.3 Results of the Design Features Review

This section presents the review results for the design features. Each standard design feature (e.g., all design features except the IPSO) was described by ABB-CE by a set of design characteristics. A full listing of the design features and their characteristics is provided in Section 18.7.1 of CESSAR-DC. During the evaluation of the characteristics, several were identified that required clarification of the wording. ABB-CE provided revised wording to satisfy the concerns of the staff. CESSAR-DC contains the latest revised version.

ABB-CE's responses to review issues cited in this report are from Reference 9 of CESSAR-DC Section 18.10, LD-93-106, Attachment 1 - Design Features Review Comment Responses, unless noted otherwise.

The seven design features were found to be acceptable. This conclusion was based upon: (1) a review of the conceptual basis of the design features as described in ABB-CE's design documentation, (2) the onsite demonstrations of the functioning of the design features on a dynamic mockup using selected EPGs and CSF success path monitoring, and (3) a top-level review of the design features using available HFE guidance. In addition, a more detailed review of the design features was conducted based on the design characteristics associated with each standard design feature and available HF guidelines. Specific design concerns were identified as a result of this review and are presented below. For each design feature, the conceptual basis of the feature as well as a review of design characteristics and HF guidance issues is provided. Some design characteristics required more extensive review and evaluation than others because their implications to overall HSI and plant performance were not fully understood initially. This review and evaluation included discussions with ABB-CE, demonstrations and examinations using the mockup, and additional reviews of supporting design documentation. The resolution of these design characteristics and the resolution of the HF guidance issues are presented in the following format: statement of characteristic or issue, discussion of evaluation, and statement of review status. All identified characteristics and issues have been resolved.

18.6.1.3.1 Evaluation of Design Features and Characteristics

18.6.1.3.1.1 DPS Display Hierarchy

The DPS display hierarchy was examined through a review of ABB-CE design basis documents and an evaluation of a mockup that provided representative screens of DPS. The mockup included the top-level IPSO display page, CFM menu, plant sector menus, and a sample set of second- and third-level display pages for selected CSFs and plant sectors. This hierarchy of displays is to be accessible from any DPS CRT in the CR. The single-point acknowledgement capability was demonstrated. This capability causes an alarm condition to be acknowledged on both the DPS and DIAS when the operator acknowledges the alarm on either system. The alarm acknowledgement and alarm message capabilities were examined. When an alarm was acknowledged, a message describing the alarm condition was observed in a spatially-dedicated alarm message window at the bottom of the DPS display. The DPS alarm list display, which is used when multiple alarms exist, was not implemented and could not be examined. The capability to acknowledge an alarm and obtain information about the alarm condition from multiple locations in the CR via the DPS were discussed.

The DPS display hierarchy provides, through operator selection, major plant status indications to support operator information requirements associated with monitoring, controlling and diagnosing plant condition. This concept addresses the operator's need to maintain awareness of significant changes in plant conditions and the implications of these changes to plant safety and operating goals. The basic concept of the DPS display hierarchy was found to adequately support this need. Final acceptance of the DPS display hierarchy will depend upon the final design implementation. ABB-CE has provided a set of 11 design characteristics associated with the DPS display hierarchy. Additional characteristics may also be considered in the post-certification review of the design, which will be performed in accordance with the ITAAC. A review of the 11 characteristics identified by ABB-CE is provided below.

18.6.1.3.1.1.1 Review of DPS Display Hierarchy - Design Characteristics

An initial review of the 11 characteristics associated with the DPS display hierarchy found the following characteristics to be acceptable - 1, 4, 5, 6, 9, and 10 - based on their support of the operator's need to access and process information regarding plant conditions. Characteristics 2, 3, 7, 8, and 11 received additional review because their implications to overall HSI and plant performance required further analysis. The results of these reviews are discussed below.

Characteristic 1: addressed the fact that the DPS display hierarchy provides access to the total set of plant data, as opposed to the subset provided by the DIAS. Plant data is organized in the DPS to support operator needs. This includes CSFs and critical success paths presented in display pages using graphical display formats such as schematic diagrams and bar charts. The display pages are organized in a three-level hierarchy with increasing levels of detail to support operator information needs for monitoring, control, and diagnosis.

Characteristic 4: each display page provides a menu window to support navigation through the display hierarchy. After an alarm has been acknowledged, a message describing the alarm condition appears in the spatially-dedicated area.

Characteristics 5 and 6: addressed the fact that alarms may be acknowledged from the relevant DPS screens and that all DPS screens can be accessed from any DPS CRT in the CR. The above characteristics support the operator's tasks by providing necessary information when it is needed.

Characteristic 9: the DPS display units can be read at the control panel. Greater viewing distances are not necessary because the full set of DPS displays can be accessed from any control panel.

Characteristic 10: the DPS display units are located on the vertical panel sections, which maintains a logical separation of displays on the vertical section and controls on the bench section of the panels.

Characteristic 2: The DPS display hierarchy provides access to displays incorporating system/component status, process parameters, and annunciator status/acknowledgement.

Evaluation: ABB-CE demonstrated available portions of the DPS display hierarchy on the mockup, including display navigation paths based on plant CSFs and plant segments and the representation of process parameters and system/component status via DPS displays. Also demonstrated were the incorporation of alarm status representations into these displays and the alarm acknowledgement capability. The incorporation of the alarms into the plant displays provides the capability to access alarm condition information and then acknowledge alarms from any DPS CRT in the CR. This characteristic provides flexibility to CR operations. IEC 964 (1.4-1) states, "An alarm shall be annunciated in the control room section where the operator has the necessary means for initiating corrective actions." The System 80+ CR provides this capability in two ways (1) the DIAS has alarm display devices that are spatially dedicated to specific control panels where the relevant controls are located, and (2) the DPS displays can also be accessed from the relevant control panels. This characteristic was found acceptable because, based on this review, it was determined that the design did provide the information defined in Characteristic 2 and the approach used for presenting this information is consistent with human factors principles for supporting operator performance.

Characteristic 3: Touch screen VDU devices are utilized.

In the DPS, user inputs are provided through touch screens. At the time of this evaluation, the touch screen of the DPS mockup was designed such that when the user touched the screen a cursor appeared above the finger tip and followed the finger tip's position. ABB-CE stated that the design is being modified to eliminate the cursor. The touch area will become backlit when touched and activated when the finger is removed, which is more typical of touch-screen implementation. Seven concerns regarding the prototype implementation were evaluated: arm fatigue, obstructed vision by arm or cursor, touch screen sensitivity, cursor positioning, alternative input methods, and the rationale for touchscreens over other technologies.

(a) Repeated input actions using the DPS may cause arm fatigue.

Evaluation: Table 3-1 of draft NUREG/CR-5908, "Advanced Human/system Interface Design Review Guideline," states that touch screens should not be used if the application requires moving/holding the arm to screen for long periods of time. While interactions with the DPS generally do not require long periods of time, their frequency may cause fatigue. This may increase the chance for operator error, slow reaction time, and reduce the operator's willingness to use the system. This was evaluated further via onsite trials. It was found that when materials are placed on the laydown space in front of the CRT the user may be required to reach toward the screen in an awkward position which can be fatiguing. While fatigue was not a serious problem during the limited duration of the onsite evaluation, it may be a concern during a full-operating shift. ABB-CE has entered Item 72 into its HF issue tracking system to provide a commitment to evaluate alternative input devices in addition to the touch screen design interface, prior to detailed design of the CR panels.

(b) The user's hand or arm may obscure some portions of the DPS screen.

Evaluation: Table 3-1 of draft NUREG/CR-5908 states that touch screens should not be used if the task will be disrupted by the hand temporarily blocking the screen. This was evaluated further via onsite trials using the mockup and found not to be a problem.

(c) The cursor, which appears above the user's finger tip, may not be visible when the user is in a seated position.

Evaluation: ABB-CE stated that the final design will not use a cursor. Instead it will use touch targets that become backlit when touched and then activated when the finger is removed. ABB-CE later demonstrated this capability on a different part of the mockup. This was found acceptable because the backlit area is visible when the user is in a seated as well as standing position.

(d) The DPS touch interface implemented in the mockup did not appear sensitive enough in that it did not always respond to touch. The touch targets appear to be susceptible to both failure to activate and inadvertent activation.

Evaluation: ABB-CE stated that this is a prototype implementation problem. DPS prototype touch screens are being improved and this problem will be corrected in the final design implementation.

(e) Positioning the cursor using touch was at times a problem due to slow response time of the cursor movement and/or difficulty positioning the cursor over the poke point.

Evaluation: ABB-CE stated that the final design will not have a cursor. Touch screen cursor response is an implementation problem with the prototype. ABB-CE design reviews will ensure that this is corrected.

(f) The design documentation does not identify any interfaces or input methods that may be available to the operator for use as alternatives to the touch screens. ABB-CE was requested to consider providing alternative input mechanisms, such as trackball, mouse, or keyboard entry to provide the operator with more that one method for providing input to the DPS and which would address the potential arm fatigue and sensitivity problems noted above.

Evaluation: ABB-CE entered Item 72 into its HF issue tracking system to provide a commitment that ABB-CE evaluate whether an alternative input device, in addition to the touch screen design, is necessary. ABB-CE stated that if an alternative is determined necessary, the feasibility of adding specific user interface devices will be considered. This evaluation will occur prior to detailed design of the CR panels.

(g) ABB-CE was requested to describe its rationale for selecting touch screens over other control/input devices, including the results of trade studies and consideration of the concerns listed in Sub-Items a through f, above.

Evaluation: ABB-CE stated that touch screens are used as an interface for accessing information (i.e., monitoring) because they support the inclination of humans to point. Trade literature was reviewed to determine the potential use of touch screen technology, to determine the capabilities of off-the-shelf touch screen products, and the availability of these products and related services. Surface acoustic wave technology was selected over designs using capacitive or resistive techniques. Infra-red touch screen technology is used on DIAS; thereby, addressing diversity concerns. A mouse or trackball is a pointing interface. Using a pointing interface requires allocation of panel real estate and employment of design techniques to address resulting seismic/missile hazard concerns. Considering these costs, the benefits of a pointing interface are considered low.

Using the touch screen interface for display screen selection, alarm acknowledgement, and information access supports the DPS monitoring interactions required for the anticipated tasks of these devices. This was confirmed by verification analysis in which touch screen response was specifically evaluated. The verification analysis recommended that touch selection could be improved by using flatter screen CRTs and improving software response. Subsequent to this analysis, new software and hardware were purchased to address this concern.

ABB-CE's response did not indicate that the selection of touch screens was derived from a systematic evaluation of human performance requirements. The response did indicate that trade studies were conducted after the initial selection of touch screens had been made to guide the selection of particular touch screen hardware/software.

Alternative input devices such as the trackball or mouse should not be excluded from consideration for use as a DPS input device on the basis of panel space requirements or seismic qualification considerations. Because trackballs can be permanently mounted to a panel they require little panel real estate. Because the DPS is not a seismically-qualified system, it is not necessary that the input device be seismically qualified. Seismic qualification of the input devices is only a concern to the extent that the DPS interfaces should have some consistency with respect to method of use with the input device of a seismically qualified system.

ABB-CE provided a commitment in Item 72 of its HF issues tracking system to evaluate alternative input devices, in addition to DPS touch screen design interface, prior to detailed design of the control panels.

Characteristic 7: The DPS automatically provides specific alarm condition messages at the time of alarm acknowledgement.

At the time of alarm acknowledgement a message appears in a message window at the bottom of the CRT screen to describe the alarm condition. The alarm message contains the parameter's descriptor, data base point ID and current value. If the parameter is in an alarm state, then the message will also contain the alarm severity and alarm set point. The concept of providing a message is consistent with draft NUREG/CR-6105, "Compilation of Alarm System Guidelines and Evaluation of Their Applicability to Hybrid and Advanced Control Rooms," guideline, "Ensure that the content of each annunciated message identifies, at a minimum, the alarm source (e.g., control power, Pump A) and the nature of the deviation (e.g., lost, failed)." The message presentation is delayed until the alarm is acknowledged. This message presentation method provides the message at the time it is needed by the operator and is consistent with HF principles for reducing visual clutter. The following specific concerns were identified with respect to the alarm messages.

(a) The presentation of alarm condition messages when multiple alarm conditions are associated with a single alarm was not clearly understood from the design documentation. ABB-CE was requested to clarify the method of presenting multiple alarm condition messages within DPS (e.g., Must the operator access the DIAS alarm list display to see all of the related alarm messages at one time?).

Evaluation: The design approach for responding to multiple alarm conditions on a single tile through the DPS was demonstrated and explained by ABB-CE. In addition, ABB-CE provided additional descriptions of the alarm system in Section 18.7.1.5 of CESSAR-DC. This section states that in the case of multiple alarms, information pertaining to the alarms may be obtained from the DPS through prioritized, hierarchical, and time- sequential alarm lists. In addition, alarm information may be obtained from the DPS unacknowledged alarm list and the DPS display pages that describe process parameters and components. This was found to be acceptable because the revised description states that alarm information can be accessed within the DPS through alarm lists displays and plant process and component displays; access to the DIAS is not required to access this information.

(b) The alarm messages do not present the operator with the required alarm response actions or references to relevant steps in the alarm response procedure. Inclusion of this information is consistent with the draft NUREG/CR-6105 guideline that states ". . . recommend that references to alarm response procedures be provided [in messages displayed on CRTs or printers]." ABB-CE was requested to clarify its position regarding inclusion of this additional information in the alarm messages.

Evaluation: ABB-CE stated that the Nuplex evolutionary design supports use of hardcopy procedures. Computer based procedures are not in the design bases because they do not represent proven technology and there is no basis for determining acceptability. ABB-CE will allocate space on the DPS CRT screen for information which will reference applicable alarm response procedures. This information would be provided by the COL applicant once alarm response procedures are developed and implemented consistent with HF standards and guidelines. This response was found acceptable.

Characteristic 8: Conforms to System 80+ Human Factors Standards, Guidelines, and Bases (HFESGB).

Evaluation: The HFESGB document was reviewed separately and found to be acceptable as a basis for the

DPS design. The results of this review are presented later in Section 18.6.3. In addition, the DPS mockup was independently evaluated against HF guidelines selected from draft NUREG/CR-5908. The results are provided below in the subsection titled Review of DPS Display Hierarchy-Design Implications. Based on the resolution of concerns identified in Review of DPS Display Hierarchy-Design Implications and Section 18.6.3 this characteristic was found to be acceptable.

Characteristic 11: The DPS display hierarchy is diverse and independent of the DIAS.

Evaluation: Diversity and independence are design considerations related to equipment reliability. The NRC I&C Branch (HICB) was requested to review this issue and provided the following evaluation:

> The DPS is physically separated and independent of both DIAS channels. Independent Class 1E power busses are provided for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, A and B instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The category 2 variables are displayed on DIAS-N and DPS with power supplies from the safety instrument buses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the C and D instrumentation bus. The redundant information systems conform to the guidelines for the physical independence of electrical systems in RG 1.75.
>
> The staff is reasonably assured that the information systems important to safety conform to the requirements of GDC 13 for monitoring systems and variables over their anticipated range for normal operation, for anticipated operational occurrences, and for accident conditions. Further, conformance to GDC 13 and the applicable guidelines satisfies the requirements of GDC 19 with respect to information systems provided in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Based on this review this characteristic was found to be acceptable.

18.6.1.3.1.1.2 Review of DPS Display Hierarchy - Design Implementation

The following ten DPS issues were identified and resolved using HFE guidelines and current design practice (subject to the detailed implementation limitations described above).

Issue 1: DPS response time

During the onsite evaluation, the time required by the DPS to respond to inputs was, at times, excessive. It was requested of ABB-CE to clarify the intended response time for the System 80+ and to identify any response time differences between the design goal and the actual performance of the mockup.

Evaluation: ABB-CE stated that transitions between screens in the DPS hierarchy were likely to require the most time while responses to other operator input actions would require less time. The reviewers timed a number of the screen changes and noted that many were at the 2-second range. This was considered slow for operator needs, given the amount of screen switching that was noted during the walkthroughs.

In its response ABB-CE stated that its application software response guideline is 2 seconds (Section 4.5 of the HFES-GB). The 2-second response time is consistent with industry guidance. This is the maximum acceptable time from the moment of touch request at the display page menu option until the directory page is displayed. This also applies to a touch request at the display page directory until the selected display page is presented. Any response time in excess of the 2 seconds is a prototype implementation problem that will be corrected on the mockup. An HFE tracking system item [Item 73] has been added to address the fact that the NRC staff found the DPS response time was slow during the onsite evaluation and has concerns that this may interfere with the operators' ability to rapidly scan and collect information.

This concern was satisfied based on ABB-CE's commitment to include this concern in the HFE tracking system.

Issue 2: RCS flow indication

The onsite review indicated that the parameter RCS flow is not available on the DPS or DIAS. The walkthroughs, which were based on critical safety factor indications, indicated that this is a potentially valuable parameter for indication/verification of proper forced cooling flow or natural circulation flow. It was requested that ABB-CE state its position regarding the inclusion of RCS flow indications in the DPS.

Evaluation: ABB-CE stated that the function and task analysis shows no use of RCS flow, RCP differential pressure, SG differential pressure, or core differential pressure to verify RCS loop circulation. Loop differential temperature, hot and cold leg temperatures, core exit thermocouples, and subcooling are required by Emergency Operations Guidelines to determine the existence of circulation. There is a low flow reactor trip on steam generator differential pressure to protect against a sheared shaft event. RCS flow (gpm,lbm/hr) is not required.

Upon further review the current set of parameters was found to be adequate based on the information requirements derived from the function and task analysis.

Issue 3: Measurement units for plant parameters are not presented

The DPS displays do not consistently show units of measurement such as temperature, pressure, level, and power for the depicted plant parameters. This is in conflict with HF design guidance including Guideline 1.-3.6-6 of the draft NUREG/CR-5908.

Evaluation: ABB-CE stated that the HFESGB document will be updated to require that measurement units will be included for all numerically-displayed values. ABB-CE entered Item 80 into the HF TOI to record its commitment to make this HFESGB modification.

Issue 4: Deviation bar chart orientation and scaling

Three concerns were identified with respect to the deviation bar charts depicted in the screen titled "Primary (level 1)" and similar DPS screens.

(a) Inconsistencies were noted in the orientation of deviation bar charts. On the comparison of charging and letdown, upward movement of the bar indicates that inventory is increasing. However, for the comparison of steam flow and feed flow, upward movement of the bar indicates that inventory is decreasing. This is potentially confusing to operators and in conflict with HF guidelines that pertain to consistency.

Evaluation: ABB-CE stated that the bar chart referred to is a mismatch bar graph not a deviation bar graph. This is a prototype implementation problem. Design review will correct this. The orientation will be changed [to be] consistent with the comment.

This change will also be reflected in ABB-CE's HFESGB document. ABB-CE committed in Item 105 of the HFE tracking system to expand the HFESGB document to include detailed guidance for display design conventions for deviation bar charts and other graphic formats. (See Section 18.6.3.3.2.1.) This concern was satisfied based on these commitments from ABB-CE.

(b) Digital values are not provided for the deviation bar charts. As a result, operators must either try to interpret values from the scale or obtain values from the DIAS. ABB-CE was asked to consider adding bands to the scales of these charts that indicate normal operating ranges to facilitate interpretation of these scales.

Evaluation: ABB-CE stated that the mismatch bar graphs are provided without digital values because they are used for qualitative display of process dynamics on Level 1 monitoring display pages. There are many power plant operating situations in which varying magnitudes and directions of mismatch (e.g., charging flow - letdown flow) are acceptable and expected. ABB-CE will consider adding a normal deviation band on a case-by-case basis for each mismatch bar graph used.

Based upon ABB-CE's explanation of the intended use of this display format, it was determined that display format was adequate and that this concern was resolved.

(c) The scales of the deviation bar charts are not scaled to facilitate reading/interpreting the displayed value. For example, the charging/letdown scale has no intermediate values between zero and 0.5m^3/min (130 gpm).

Evaluation: ABB-CE agreed to modify Section 18.5.1.5.3 of CESSAR-DC to require that parametric requirements for display and control variables be defined in terms of precision, in addition to device type, range, accuracy, and units of measurement, as part of the task analysis methodology. ABB-CE also committed to modifying Section 6.1.5.2 - Phase 2 Availability Inspection Criteria of its verification and validation plan (NPX80-IC-VP790-03) to indicate that precision specifications will be verified for each as-built item of the HSI. In addition, ABB-CE agreed to enter Item 105 into the TOI to record its commitment to provide additional guidance in the HFESGB regarding graphic formats such as mismatch and deviation bar charts.

Issue 5: Bar chart scaling

Three concerns were identified with respect to normal bar charts as depicted in the screen titled "RCS Control (PRI) Level 2" and similar DPS screens.

(a) Digital values are not provided for the normal bar charts. As a result operators must either try to interpret values from the scale or obtain values from the DIAS.

Evaluation: ABB-CE stated that the bar charts referred to are on Level 2 displays (RCS control and inventory control). The bar charts display the parameters which comprise the Level 1 monitoring display's mismatch bar graph (i.e., charging flow and letdown flow). The bar charts on the Level 2 control displays provide a finer degree of information (the scales are demarcated in increments of $0.04 m^3/min$ (10 gpm) but the precision afforded by digital values is not required at the bar chart. The precision is required and provided at the ultimately controlled parameter (i.e., pressurizer level) which is also located on the same display page. That is, the precision is provided on the RCS parameters because it is the RCS control page. On the CVCS control page, digital values are provide for charging flow and letdown flow because it is this page that is designed to be used for monitoring when controlling CVCS components and process parameters. Also digital values and finer scales are provided on the process control devices where these parameters are controlled.
Based upon ABB-CE's explanation of the intended use of this display format, it was determined that display format was adequate and that this concern was resolved.

(b) Units of measurement (e.g., m^3/min (gpm)) are not provided. This is in conflict with HF design guidance including Guideline 1.3.6-6 of draft NUREG/CR-5908.

Evaluation: This issue is covered by the response to Issue 3 above. Units of measurement will be provided.

(c) The scales of the bar charts are not scaled to facilitate reading/interpreting the displayed value. The scale is numbered zero, 100, and 200. Lines are provided to indicate 50 and 150.

Evaluation: ABB-CE stated that this is a prototype implementation concern that will be corrected by ABB-CE design review.

Issue 6: Labeling of symbols

Symbols used to represent major plant components in the DPS displays are not consistently labelled. For example, in the screen titled "Primary (PRI) Level 1" the labels for symbols for reactor coolant pumps, steam generators, and pressurizer are located below the symbol. In other DPS screens, the label appears inside the symbol. This is in conflict with HF guidelines for consistency and symbol labeling.

Evaluation: ABB-CE stated that this is a prototype implementation concern. Any labeling that does not meet our guidelines to be meaningful, unambiguous, consistent, compatible, take into consideration the users, their tasks, working environment, and specific guidance for data descriptors (HFESGB Sections 2.1.b, 2.1.c, 2.1.d, 2.1.e, 2.1.h and 4.1.5) will be corrected by ABB-CE design review. Based on this guidance, it is ABB-CE's assessment that labels in two locations for specific reasons provide no hinderance in using the display and introduces no likely human errors even though every label position is not identically oriented.

Upon further consideration ABB-CE's position that this was an acceptable deviation from human factors guidance was found acceptable. This concern is satisfied.

Issue 7: Improper terminology

The display pressurizer pressure (PRI) Level 3 uses obsolete terminology - SDS.

Evaluation: ABB-CE stated that this a prototype implementation problem. The design and the prototype will be made consistent with system design. Display terminology will be corrected by design review.

Issue 8: Poke point labeling

In the display pressurizer pressure (PRI) Level 3, single characters such as T or P are used as labels for dynamic parameters. These parameters are coded in cyan to draw attention to the fact that they are dynamic. These labels are also used to designate poke points for obtaining more detailed data about the particular parameter. Two concerns regarding the small size of these labels were identified.

(a) The convention of coding these labels with the color cyan loses its effectiveness when short labels are used; the ability to discern color decreases with target size.

Evaluation: ABB-CE stated that the issue of single letter designators is a trade off between screen clutter and identification. The present use of cyan single letter parameter labels has been demonstrated to be effective [during suitability analysis]. Single letters are used only on these most frequently used labels (P = pressure, T = temperature, F = flow, L = level) to reduce screen clutter. This meets the reference general guidance (HFES-GB Section 2.1).

(b) The size of the single letter label violates guidelines for minimum poke point size.

Evaluation: ABB-CE indicated that the poke area size is larger than the single letter designator. Section 3.4.9.1 of the HFESGB specifies a size of at least 1.6 cm^2 (.25 square in.) in area and 0.64 cm (.25 in.) in height. ABB-CE further stated that the parameter descriptor is smaller and centered within the poke area. The minimum poke area is defined to assure that the finger will not address more than one area when the finger is directed at the parameter or component descriptor. Poke areas do not overlap.
This concern was resolved based on ABB-CE's explanation that the actual poke area is larger than the single letter designator.

Issue 9: Operator aids

In general the DPS does not provide adequate operator aids such as a RCP seal diagnosis chart or a display correlating pressurizer level with RCS volume. Inclusion of such aids may enhance operator performance by making effective use of the processing capabilities that computer-based information systems possess. It was requested that ABB-CE clarify its position regarding the inclusion of computer-based operator aids in the DPS.

Evaluation: ABB-CE stated that this suggestion has been added to the TOI system [item 86]. ABB-CE's position on computer-based operator aids is that they are a good idea where they actually enhance the operator's ability to perform anticipated tasks. Evaluation of the benefits and drawbacks of each potential operator aid is warranted. ABB-CE relies on the results of a multi-disciplinary design review, procedure guidelines, and utility feedback for potential operator-aid candidates for the Nuplex 80+ standard design. ABB-CE will add operator aids to the Nuplex 80+ control room design contingent upon the results of the evaluation. The evaluation will identify useful operator aids but avoid those which may be distracting for the operators.

Issue 10: CRT glare

A potential problem with CRT glare was noted during the onsite review. There was a concern that the ambient lighting in the mockup room is dimmer than the anticipated CR conditions. In addition, the selection of CRTs that are different from those used in the mockup may increase glare. It was requested that ABB-CE to describe the measures that will be taken to reduce glare in the final design.

Evaluation: ABB-CE stated that its design criteria (HFESGB Section 3.4.6) recognizes the need to reduce CRT glare. ABB-CE stated that it will continue to evaluate CRT hardware for its acceptability in reducing CRT glare and that the concern will be resolved by design review. In addition, evaluation of CRT glare characteristics is explicitly included in the staff's review of the final design as per NUREG-0700 and HFE PRM Element 8, "Verification and Validation."

18.6.1.3.1.2 DIAS Alarm Tile Display

The onsite review examined the DIAS alarm tile displays that were resident on the RCS panel. A set of DIAS alarm tiles were also examined at the CVCS panel. The DIAS alarm tiles contained groups of alarms that were functionally related to each other and to the RCS panel. The alarm tiles were spatially dedicated within the display page. The DIAS alarm tiles were presented on electro-luminescent panels on the vertical section of the RCS panel.

The DIAS alarm tile display system is coordinated with the DPS display system such that (1) the same coding schemes are used in the DIAS and DPS for indicating alarm priority and status, (2) similar alarm messages appear in both the DIAS and DPS message windows (DPS messages are more detailed), and (3) alarms that are acknowledged by the operator on one system are also acknowledged on the other system.

The DIAS alarm tile display system is an operator-alerting system that conveys the meaning and importance of alarm conditions through a hierarchical classification of alarm conditions and spatial dedication of alarm messages. This concept was found to be acceptable based on current alarm system guidelines and research addressing the value of alarm message prioritization/filteringand spatial dedication as techniques for reducing operator workload associated with handling alarm messages. Specific concerns are reviewed below.

18.6.1.3.1.2.1 Review of the DIAS Alarm Tile Display - Design Characteristics

An initial review of the 17 characteristics (see Section 18.7.1 of CESSAR-DC,) associated with the DIAS alarm tile display found the following characteristics to be acceptable: 1, 6, 7, 8, and 10. Characterisitics 2, 3, 4, 5, 9, 11 through 17 required additional review. Characteristics 1, 6, 7, 8, and 10 as described in the design literature and represented in the mockup, were found to be generally consistent with accepted HF design principles and guidelines related to providing necessary information to support operator tasks. Some apparent discrepancies are noted below and were resolved through commitments for further review when the design is more complete.

Characteristics 1, 6, and 8 addressed the fact that individual alarm messages are organized using functional grouping and spatial dedication. A single alarm tile may be associated with a group of individual alarm conditions that are functionally related (characteristic 1). Alarm tiles are further grouped by functional relationships within an alarm tile display page (characteristic 8). An alarm tile display device is assigned to a control panel based on functional relationships with the indicators and controls that reside on that panel (characteristic 6). These characteristics support the operator's interpretation of the alarm message content; much of the alarm message content is conveyed by the location of the tile in the CR.

Characteristic 10 addressed the fact that while the alarm tile messages must be read at the panel, their status is visible from across the controlling work space. This characteristic supports operator awareness of plant condition and is consistent with NUREG-0700, which states that an alarm should be readable from the position in the work station where the annunciator will be acknowledged.

The DIAS alarm tile display units are located on the vertical panel sections (characteristic 7), which maintains a logical separation of displays on the vertical section and controls on the bench section of the panels.

Characteristics 2, 3, 4, 5, 9, 11, 12, 13, 14, 15, 16, and 17 received additional review because their implications to overall HSI and plant performance required further analysis. The results are discussed below.

Characteristic 2: Touch screen VDU devices are utilized. In the DIAS alarm tile display, user inputs are provided through touch screens. Table 3-1 of draft NUREG/CR-5908 states that touch screens should not be used if the application requires moving/holding the arm to screen for long periods of time or if the task will be disrupted by the user's hand temporarily blocking the screen.

Evaluation: These concerns were evaluated through use of the mockup. Arm fatigue was not determined to be a problem because use of the DIAS required momentary touches by the operator; it did not require the operator to hold the finger over the touch area for a long period of time. The alarm tiles were touched repeatedly in a manner that simulated acknowledgement of multiple alarms. Arm fatigue was not noted after repeated touches. Use of the touch screen did cause the finger and portions of the hand to block the operator's view of portions of the alarm tile display. However, this did not interfere with the operator's task of acknowledging the alarms because portions were blocked momentarily, only. This characteristic was found to be acceptable because arm fatigue and blockage of the alarm tile display were not found to be problems.

Characteristic 3: On each DIAS alarm tile display device, the status of alarm tiles is presented on a single alarm tile display page; for each tile, an associated alarm list page is available to present the status of individual alarm conditions.

Evaluation: The DIAS alarm tile display uses an alarm tile format to convey alarm status information such as priority and state for a specific set of plant parameters. An alarm message list format is used to convey information about the specific alarm conditions that were responsible for generating the alarm. This concept of using alarm tile and list formats together to convey plant status is consistent with Guideline 3.1-3 of draft NUREG/CR-6105, which states that for computerized annunciator systems, consideration should be given to including a mix of VDT-displayed warnings and tile-displayed warnings. Therefore, this characteristic was found to be acceptable.

Characteristic 4: Unacknowledged alarms on a single tile are acknowledged through the display as a group.

Evaluation: This characteristic is consistent with traditional CR designs in which groups of alarms are acknowledged through a single operator action. In traditional control rooms each alarm condition generally has a dedicated alarm tile and pushing the acknowledge button acknowledges the active alarms. In the DIAS, an alarm tile may have a group of alarms associated with it. When an alarm tile is acknowledged detailed information (e.g., title, parameter, setpoint) related to the individual associated alarms is presented on the DIAS alarm tile list display page. Therefore, this characteristic was found to be acceptable because it operates in a manner similar to traditional alarm systems but organizes alarms in functionally related groups which may provide additional benefit to the operator's understanding of plant conditions.

Characteristic 5: Alarm condition messages are automatically provided upon alarm tile acknowledgement.

The concept of providing a message is consistent with Guideline 3.3.3-3 of draft NUREG/CR-6105, which states that the content of each annunciated message should identify, at a minimum, the alarm source (e.g., control power, Pump A) and the nature of the deviation (e.g., lost, failed). The message presentation method of the DIAS alarm system provides the message at the time it is needed by the operator and is consistent with HF guidelines for reducing visual clutter. Upon acknowledgement of an alarm tile a message appears at the bottom of the display device to describe the alarm condition. If multiple alarm conditions were associated with the alarm tile then the DIAS alarm list display will automatically be presented. However, the following three concerns have been identified.

(a) When the alarm list display is presented it replaces the alarm tile display and the operator can no longer see the alarm tiles. It is understood that the alarm list display shows an "Alarm Stat" tile, which indicates the status of highest level active alarm from the alarm tile page. However, it appears that this tile does not show the title of the highest priority alarm, only its status. Also, the other alarm tiles that may have similar or lower levels of priority are not visible at this time. There is a concern that while the alarm list display is shown, the operator may lose site of other alarms. Under what conditions is the alarm tile list display page returned again to the DIAS alarm tile display (e.g., new alarm condition, operator action, automatic time-out)?

Evaluation: The alarm tile list display page returns to the DIAS alarm tile display upon operator action, only. The design of the DIAS alarm list display page includes the alarm tile of interest and an alarm 'STAT' tile. The operation of this characteristic was demonstrated to the review team using the mockup and found to be acceptable because the 'STAT' tile provides a salient alert to the operator regarding the presence of new alarms. In addition, the ability of operators to use the 'STAT' tile and not lose cognizance of other alarms on that alarm tile matrix while the alarm list is displayed will be evaluated during the validation tests of the completed design.

(b) It does not appear that alarm set points are displayed on:

- The message window of the alarm tile display, or
- The alarm list display if the tile has not yet alarmed

During the onsite review, ABB-CE was requested to clarify its position on presenting alarm setpoint information via DIAS for parameters that are not in alarm conditions.

Evaluation: ABB-CE stated that it will add design guidance relating to alarm setpoints contingent upon the results of a multi-disciplinary design review. This concern has been added to the TOI system [item 74].

This concern was satisfied through ABB-CE's inclusion of this concern in its HFE tracking system.

(c) The alarm messages do not present the operator with the required alarm response actions or references to relevant steps in the alarm response procedure. Inclusion of this information is consistent with Guideline 3.3.1-5 of draft NUREG/CR-6105, which recommends that references to alarm response procedures be provided in messages displayed on CRTs or printers and that the document title, major section, and page numbers are included in such references. ABB-CE was asked to further describe the operation of the alarm tile and list displays and clarify its position regarding inclusion of this additional information in the alarm messages.

Evaluation: ABB-CE did not indicate plans to provide this capability on the DIAS but did commit to provide additional space on the display pages of the DPS to allow the inclusion of references to applicable alarm response procedures. This information would be provided by the COL applicant once alarm response procedures are developed and implemented. The intent of this concern is addressed satisfactorily by the inclusion of this capability on the DPS rather than the DIAS because this information will be readily available to the operator via the DPS.

Characteristic 9: DIAS alarm tile displays are configured to conform to System 80+ human factors standards, guidelines, and bases.

Evaluation: This characteristic was found to be acceptable because the HFESGB document was reviewed separately and found to be generally acceptable as a basis for the HSI design for the System 80+.

Characteristic 11: Alarm tiles are established for process parameters that provide direct indication of:

(a) critical safety functions
(b) critical power production functions
(c) success path performance
(d) success path availability
(e) damage to major equipment
(f) personnel hazard

Evaluation: Guideline 1.2-2 of draft NUREG/CR-6105 states that to warrant inclusion in the alarm system, a potential alarm source should require operator action to stabilize deviant conditions or to verify automatic control equipment conditions. In addition, the deviancy should be such that normal surveillance activities of personnel cannot be relied on to result in its reliable detection within acceptable time periods. Guideline 1.2-6 of draft NUREG-/CR-6105 states that, where practical, alarms should be provided such that the operator is alerted before a major system or component problem results in a condition which causes a loss of availability (e.g., reactor trip), equipment damage, violation of Technical Specifications, or other serious consequences. The six alarm categories (a through f) are consistent with these guidelines.

The NRC Instrument and Controls Branch (HICB) reviewed this concern further. The staff stated that it was reasonably assured that the information systems that are important to safety conform to the requirements of GDC 13 for operation, for anticipated operational occurrences, and for accident conditions.

The staff evaluated the plant process display instrumentation and has found the instrumentation to be acceptable. The staff's evaluation was presented in Section 7.5.2 of the DSER. The open items in Section 7.5 of the DSER have been resolved. Therefore, this characteristic was found to be acceptable.

Characteristic 12: Alarms are presented in one of four alarm states: new, existing, cleared, and reset.

Evaluation: The use of four alarm states is consistent with Guideline 3.2.2-1 of draft NUREG/CR-6105, which states that the annunciation sequence for each alarm should uniquely indicate: incoming alarms, acknowledged alarms, and cleared alarms. The use of coding schemes (flashing, intensity, and audible tones) to designate these alarm states are generally consistent with the following guidelines from draft NUREG/CR-6105:

- 3.2.2-1 (flashing, audible, and other visual coding)
- 3.2.2-2 (audible and other visual coding)
- 3.2.2-3 (audible coding)
- 3.2.2-4 (flashing and color coding)

The specific flash rates used in the implementation of codes for alarm state are a concern. ABB-CE stated that new alarms would have a 1-second flash cycle with a 50-percent ON duty cycle (i.e., ON for 0.5 seconds and then OFF for 0.5 seconds) and the cleared alarms would have a 2-second cycle with a 25-percent ON duty cycle (i.e., ON for 0.5 seconds and then OFF for 1.5 seconds). The resulting flash rates for new and cleared alarms are 1 and 0.5 Hz, respectively. These flash rates are slower than the flash rates recommended by Guideline 1.3.10-13 of draft NUREG/CR-5908, which states, "A flash rate in the range of 2 to 5 Hz, with a minimum duty cycle (On interval) of 50 percent should be used." These flash rates are also slower than Guideline 2.3.3.3 of ABB-CE's HFESGB, which indicates that when two flash rates are used the higher priority state shall be between 3 and 5 Hz and the lower priority state shall be between 1 to 2 Hz. ABB-CE stated that it will implement flash rates that are consistent with the HFESGB guidance while meeting the functional design requirement to show unacknowledged alarms simultaneously with the highest priority existing alarm.

The following additional concerns were identified as a result of a demonstration of the alarm tile display system during the onsite review on May 13 and 14, 1993:

- The flash rates for new and cleared alarms did not appear to be sufficiently different to facilitate rapid discrimination.
- Due to the timing of flash rates of new and cleared alarms the illuminated alarm tiles produced the illusion of motion between adjacent alarm tiles (marquee effect). There was a concern that this would be visually distracting to operators.
- While the alarm tile display concept has many characteristics that are individually consistent with current HF guidelines, the integration of these characteristics represents an innovation in alarm presentation that is largely untested. The effectiveness of this presentation method as an integrated part of a human/system interface cannot be predicted from past experience or current research. This concern extends beyond issues of perception and discrimination of individual tiles. It includes the ability of operators to maintain awareness of plant status and extract necessary information in coordination with plant dynamics.

ABB-CE agreed to review each of these concerns further and entered Items 75, 76, and 77 into the TOI to record its commitment. In addition, ABB-CE subsequently agreed to evaluate these and other issues related to the alarm system using a stand-alone DIAS alarm tile display prototype, prior to verification and validation. ABB-CE entered Item 101 into the TOI to record its commitment to perform this evaluation. Based on these commitments this characteristic was found to be acceptable.

Characteristic 13: Individual alarm tiles have the capability to indicate either the highest priority of new or cleared alarm (i.e., N1, N2, N3, C1, C2, C3 in that order of priority) while continuing to indicate the highest priority existing alarm.

Evaluation: This characteristic refers to the fact that when alarm tiles flash for new or cleared alarms, the highest priority existing alarm is also visible between flashes. The concept of presenting the highest priority conditions is consistent with Guideline 1.1-5 of draft NUREG/CR-6105, which states that the alarm system should have display functions to permit the operator to easily identify an alarm and its seriousness. In addition, Guideline 1.4-1 states that the steadied (acknowledged) alarm shall be indicated to ensure that its existence is not forgotten. Guideline 1.1-3 of draft NUREG/CR-6105 states that the system should reduce the overall number of discrete visual and aural alerts. This concept reduces the number of alarm tiles that the operator is presented but does not limit the operator's access to alarm information because all alarm information is available elsewhere in the CR.

Alarm priority is coded by shape. This is consistent with draft NUREG/CR-6105 Guideline 3.2.1-1 which states that acceptable methods for priority coding include color, position, shape, or symbol coding. The implementation of this characteristic, including simultaneous presentation of multiple alarm states on a single tile, was reviewed at the mockup. When presented on a single alarm tile the alarm presentation scheme was considered adequate by reviewers. However, when this scheme was presented on multiple alarm tiles it was noted that the flashing of adjacent alarm tiles interact in undesirable ways (e.g., when new and clearing alarm tiles conditions exist on adjacent tiles, the alarm tile pattern appears to move from one tile to the other). (See the review of characteristic 12, above.)

ABB-CE agreed to consider this concern during subsequent suitability verification, design review, and validation. ABB-CE entered Item 76 into the TOI system to record its commitment. In addition, ABB-CE subsequently agreed to evaluate this and other issues related to the alarm system using a stand-alone DIAS alarm tile display prototype, prior to verification and validation. ABB-CE entered Item 101 into the TOI to record its commitment to performing this evaluation. Based on these commitments this characteristic was found to be acceptable.

Characteristic 14: An alarm tile stop-flash capability is provided for use during situations of high alarm activity to focus attention on new priority 1 alarms by temporarily stopping the flashing of all other unacknowledged alarm states.

Evaluation: This characteristic is consistent with Guideline 1.3.10-14, Flash Suppression, of draft NUREG/CR-5908, which states that event acknowledgement of flash suppression keys should be provided. On this basis this characteristic was found acceptable.

Characteristic 15: A momentary tone provides an initial audible alert of the transition of one or more alarms to new or cleared states for priority 1 or 2 alarms.

Evaluation: The concept of using an audible alarm to indicate new and cleared alarms is consistent with draft NUREG/CR-6105 Guideline 3.2.2-1, which states that the annunciator sequence for each alarm should uniquely indicate incoming alarms, both by visual (e.g., flashing) and audible means; and clearing alarms, by visual and/or audible means, if the operator is required to take action on alarm clearing. The concept of using only an initial tone rather than a continuous tone is consistent with draft NUREG/CR-6105 Guideline 1.12-6 which states that the alarm system shall be designed to minimize distractions and unnecessary workload placed on the operators by the alarm systems. Based on these criteria this characteristic is acceptable.

Characteristic 16: A momentary reminder tone provides a recurring audible alert if priority 1 or 2 alarms remain unacknowledged.

Evaluation: The concept of a reminder tone rather than a continuous tone is consistent with draft NUREG/CR-6105 Guideline 1.12-6 which states that the alarm system shall be designed to minimize distractions and unnecessary workload placed on the operators by the alarm systems. The presence of a reminder tone, as compared to no reminder tone, reduces the burden on the operator's memory and is consistent with draft NUREG/CR-6105 Guideline 1.1-3 which states that the alarm system should reduce the demands on operator memory requirements and operator decision-making requirements.

The reminder tone was implemented with an activation interval of one minute. It was requested that ABB-CE provide a basis for the selection of this interval. ABB-CE stated that the one-minute interval was based on a one-minute task time per ANS 58.8, "Proposed Criteria for Safety Related Operator Actions." (ANS 58.8 uses the value of one minute as a conservative estimate of typical operator tasks. Therefore, the alarm reminder interval is roughly equivalent to the duration of other tasks that the operator may be performing.) The objective of the interval is to remind without annoying or becoming excessively intrusive. ABB-CE stated that this will be confirmed in integrated operation during validation. The reminder interval was found acceptable based on ABB-CE's rationale of operator task time.

Characteristic 17: Alarm tones emit from the console where the alarm display is located.

Evaluation: Draft NUREG/CR-6105 Guideline 3.4.4-16 recommends that each major console be equipped with a separate sound generator capable of producing a distinctive sound. This guideline supports the use of tone generators that emit from the console where the alarm display is located. However, it recommends that different tones be used for each panel. The documentation provided by ABB-CE does not indicate whether the same or different tones will be used. Requested ABB-CE to clarify its position on this matter.

ABB-CE stated that the same new alarm tone, cleared alarm tone, and unacknowledged alarm "reminder" tones are used on all (3) consoles because:

- The spatial tone direction without further distinguishing features is acceptable for the desired reliance on DIAS alarm tile acknowledgement.
- This eliminates potential confusion by limiting audible alerts to only three tones for control panel alarms (new, cleared, reminder).
- Alarm location is shown on the DPS display page menu options by plant sector and display page directories which are organized by panel.
- The alarm tile is displayed, accessed, and acknowledged on any CRT. The operator does not have to leave the console because alarm tile acknowledgment or alarm condition diagnosis can be accomplished at any control panel CRT.

The use of the same new alarm tone, cleared alarm tone, and unacknowledged alarm "reminder" tones for all (3) the consoles was found acceptable because alarm information was readily accessible from any DPS CRT, therefore reducing the need to direct the operator to specific consoles via a unique tone.

18.6.1.3.1.3 DIAS Dedicated Parameter Display

The onsite review examined the DIAS dedicated parameter displays that were present on the RCS and CVCS panels. Each dedicated display continuously shows a value that has been derived from a set of redundant sensors. A lower-level display page may also be accessed from the dedicated parameter display to view redundant sensor values as well as the derived validated value. The DIAS dedicated parameter displays were spatially dedicated within the RCS panel. The current digital values of the plant parameters are displayed in large characters allowing them to be read from across the main control console. Display details such as trends and redundant sensor data must be read at the control panel.

The DIAS dedicated parameter displays are a set of continuously-available indications of key plant parameters that are located in fixed, spatially-dedicated positions in the CR. These locations are coordinated with the locations of other functionally-related indicators and controls. This concept is consistent with HF guidelines that address the use of spatial dedication and functional grouping as a way of organizing displayed data. It is also consistent with Guideline 1.1-22 of draft NUREG/CR-5908 which states, "Dedicated displays should be available to provide continuous indications of a minimum set of parameters necessary to assess the safety status of the plant." (This guideline was derived from SPDS requirements).

18.6.1.3.1.3.1 Review of the DIAS Dedicated Parameter Display - Design Characteristics

An initial review of the 12 characteristics (see Section 18.7.1 of CESSAR-DC), associated with the DIAS dedicated parameter display found the following characteristics to be acceptable: 1, 2, 3, 4, 7, 8, and 10. Characterisitics 5,6,9, and 11 received additional review. Characteristics 1 and 2 address the fact that the values from redundant sensors are processed and a validated value is presented on the dedicated parameter display. The display system provides indications to the operator when significant discrepancies are identified between the redundant sensor values and provides indication of the quality of the displayed value. The operator is provided with the capability to examine the individual sensor values and select the individual sensors to be used for calculating the dedicated parameter value. These characteristics are consistent with Guidelines 1.4.1-9 and 1.4.1-11 of draft NUREG/CR-5908. Characteristics 3, 7, and 8 pertain to the locations of the dedicated parameter displays in the CR. Characteristic 3 is consistent with HF guidelines that address the use of spatial dedication as a mechanism for organizing data presentation and characteristic 7 is consistent with HF guidelines that require related controls and displays to be located together. Characteristic 8 indicates that the displays are located on the vertical section of the panel which maintains the separation of controls on the bench section and displays on the vertical section of the control panels.

Characteristic 10, which indicates that the dedicated parameter displays can be read from across the main control console, is consistent with HF guidelines that state that display character size should be consistent with the anticipated viewing distances of the operator.

Characteristics 5, 6, 9, and 11 received additional review because their implications to overall HSI and plant performance required further analysis.

Characteristic 5: DIAS dedicated parameter displays incorporate automatic range change features.

Evaluation: This characteristic is intended to facilitate the monitoring of multiple indicators with different ranges by providing a single indicator that is capable of changing its displayed range in response to plant conditions. This characteristic was found acceptable based on its implications for reduced operator workload.

Characteristic 6: Touch Screen VDU devices are utilized.

Evaluation: Table 3-1 of draft NUREG/CR-5908 states that touch screens should not be used if the application requires moving/holding the arm to the screen for long periods of time or if the task will be disrupted by the user's hand temporarily blocking the screen. The following concerns were evaluated during the onsite evaluation: arm/hand fatigue, arm/hand obscuring the display, and acceptability of poke point size. This characteristic was found to be acceptable because, based on the on-site evaluation, the touch screens are not anticipated to be a source of arm/hand fatigue, visual blocking of the display was momentary and is not anticipate to interfere with operator tasks, and the poke point size was consistent with human factors guidelines.

Characteristic 9: DIAS dedicated parameter displays are configured to conform to the System 80+ human factors standards, guidelines, and bases.

Evaluation: The HFESGB document was reviewed separately and found to be acceptable as a basis for the HSI design.

Characteristic 11: DIAS dedicated parameter displays are provided for the following

- critical safety functions
- success path performance
- post-accident monitoring instrumentation (PAMI) indication
- RG 1.97, "Instrumentation for LWR Nuclear Power Plants to Assess Plant and Environs During and Following Accident"

Evaluation: An effective method of supporting rapid comprehension of plant status is the use of spatially-dedicated, continuously-presented plant displays for key plant parameters. The CSFs, success path performance, and RG 1.97/PAMI, are key parameters that are related to safety. Due to their importance to plant safety they are acceptable parameters for the DIAS dedicated parameter displays. This is consistent with Guideline 1.1-22 of draft NUREG/CR-5908, which states, "Dedicated displays should be available to provide continuous indications of a minimum set of parameters necessary to assess the safety status of the plant."

The NRC HICB was requested to review this concern further. The staff evaluated the plant process display instrumentation and found the instrumentation to be acceptable. The staff's evaluation was presented in Section 7.5.2 of the DSER. The open items in Section 7.5 of the DSER have been resolved.

Characteristic 12: DIAS dedicated parameter displays are diverse and independent of the DPS display system.

Evaluation: HICB was requested to review the above characteristic. The staff stated that the DPS is physically separated and independent of both DIAS channels. Independent Class 1E power buses are provided for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, A and B instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The Category 2 variables are displayed on DIAS-N and DPS with power supplies from the non-safety-related instrument buses and computer bus, respectively. Both are battery-backed. The instrument channels are powered from the C and D instrumentation bus. The redundant information systems conform to the guidelines [NRC staff guidance] for the physical independence of electrical system in RG 1.75.

Based on this review this characteristic was found to be acceptable.

18.6.1.3.1.3.2 Review of Dedicated Parameter Display - Design Implementation

Selected DIAS dedicated parameter displays were examined and evaluated against HF guidance and current design practices. The following two issues were identified and resolved.

Issue: Resolution of trend displays.

The vertical resolution of trend displays such as pressurizer pressure and level is quite small and may not be adequate for monitoring purposes. The normal operations band is narrow and the operator may have difficulty determining whether the current value is trending toward a limit. During the onsite evaluation ABB-CE stated that the range of the scales is not adjustable (i.e., operators cannot change the limits of the display such that a desired range of the scale is shown with greater resolution). Requested ABB-CE to describe its position regarding adjustable trend displays and any implications to other DIAS capabilities, such as the automatic range change features of the dedicated parameter displays.

Evaluation: During the onsite review, ABB-CE described its position regarding this display. ABB-CE stated that it is not intended that the DIAS displays be used to determine if values are trending towards limits. The DIAS control bands are only provided as reference points for the normal operating range. ABB-CE further explained that this display presents trends with the same resolution as the strip chart recorder that it replaces. These displays are provided to present continuously visible status histories for selected parameters that are controlled automatically. If the operator requires trends with different resolutions than the dedicated parameter displays, then these may be obtained from the DPS. The DPS provides the capability to present adjustable trend plots for a broad range of plant parameters. The trend display resolution was found acceptable based on ABB-CE's description of the intended use of this display and because the DPS will have the capability of providing adjustable trend plots for a broad range of plant parameters.

Issue: Trending time intervals.

DIAS provides trends for selected variables over a 30-minute period. This interval may not be appropriate for all variables in all situations. Guideline 1.2.5-7 of draft NUREG/CR-5908 states that trend displays should be capable of displaying trends over a variety of time intervals to address operator information needs. Requested ABB-CE to address the issue of providing the trend displays with the capability of trending data over a variety of time intervals.

Evaluation: ABB-CE stated that the various DIAS trend displays were implemented in the prototype with the same trend time and, the capability exists and will be used to specify trend times based on the specific use of the trend displays. DIAS dedicated parameter displays are provided for post accident parameter monitoring, so the EOGs will define their trend times. The operator always has the DPS capability to establish ad hoc trends with adjustable trend times.

This was found acceptable based on ABB-CE's explanation that the trends will be established based on the EOGs.

18.6.1.3.1.4 DIAS Multiple Parameter Display

The DIAS multiple parameter display is a spatially-dedicated device that displays a limited set of functionally-related plant parameters on multiple display pages. The plant parameters are a redundant subset of those presented by the DPS. However, the multiple parameter display does not display data trends. Each DIAS multiple parameter display device contains a group of functionally-related parameters that are accessed through operator action. The display set for each parameter includes analog/menu pages and system parameter selection pages. The analog/menu pages contain bar graphs of the selected parameters and menus for display navigation. The system parameter selection pages provide buttons for system value selection. The onsite review examined the DIAS multiple parameter displays that were present on the RCS and CVCS panels.

The DIAS multiple parameter displays are display devices that contain a selection of functionally-related plant parameter displays and are spatially dedicated in the CR with respect to other functionally-related displays and controls. This is consistent with HF guidelines pertaining to functional grouping and spatial dedication to support operator monitoring tasks. The use of a computer-based medium for selecting and displaying individual parameters as consistent with current design practice for other computer-based plant display systems such as the SPDS.

18.6.1.3.1.4.1 Review of the DIAS Multiple Parameter Display - Design Characteristics

An initial review of the 10 characteristics (see Section 18.7.1 of CESSAR-DC,) associated with the DIAS dedicated parameter display found the following to be acceptable: 1, 2, 3, 4, 6, 7, and 9. Characterisitics 5, 8, and 10 required additional review. Characteristic 2 addresses the fact that the values from redundant sensors are processed and a validated value is presented on the parameter display. The display system provides indications to the operator when significant discrepancies are identified between the redundant sensor values and provides indication of the quality of the displayed value. This use of redundant sensors and indications of data quality is consistent with Guidelines 1.4.1-9 and 1.4.1-11 of draft NUREG/CR-5908.

Characteristics 1, 3, and 4 pertain to the fact that once a parameter has been selected by the operator it is continuously presented in analog and digital formats. These characteristics are generally consistent with HF guidelines pertaining to the presentation of data to support operator task requirements. In addition, the data presented was consistent with the information requirements of the limited set of operator tasks that were addressed during the walkthroughs of the onsite review.

Characteristic 6 refers to the fact that plant parameters are combined into individual multiple parameter display units and assigned to panels based on plant system relationships. This is consistent with HF guidelines pertaining to functional grouping of information. Characteristic 7 indicates that the displays are located on the vertical section of the panel which maintains the separation of controls on the bench section and displays on the vertical section of the control panels.

Characteristic 9, which indicates that the dedicated parameter displays are read at the panel, is consistent with HF guidelines that state that display character size should be consistent with the anticipated viewing distances of the operator.

Characteristics 5, 8, and 10 received additional review because their implications to overall HSI and plant performance required further analysis. The results are discussed below.

Characteristic 5: Touch screen VDU devices are utilized.

Evaluation: Table 3-1 of draft NUREG/CR-5908 states that touch screens should not be used if the application requires moving/holding the arm to the screen for long periods of time or if the task will be disrupted by the user's hand temporarily blocking the screen. The following concerns were evaluated during the onsite evaluation: arm/hand fatigue and arm/hand obscuring the display. This characteristic was found to be acceptable because, based on the on-site evaluation, the touch screens are not anticipated to be a source of arm fatigue and the visual blocking of the display by the hand/arm was momentary and is not anticipated to interfere with operator tasks.

Characteristic 8: DIAS multiple parameter displays are configured to conform to the System 80+ human factors standards, guidelines, and bases.

Evaluation: The HFESGB document was reviewed separately and found to be acceptable as a basis for the HSI design.

Characteristic 10: DIAS multiple parameter displays are diverse and independent of the DPS display system.

Evaluation: HICB was requested to review characteristic 10. The staff stated that the DPS is physically separated and independent of both DIAS channels. Independent Class 1E power buses are provided for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, A and B instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The Category 2 variable are displayed on DIAS-N and DPS with power supplies from the safety-related instrument buses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the A and B instrumentation bus. The redundant information systems conform to the guidelines for the physical independence of electrical system in RG 1.75.

Based on this review this characteristic was found to be acceptable.

18.6.1.3.1.4.2 Review of DIAS Multiple Parameter Display - Design Implementation

Selected DIAS multiple parameter displays were examined and evaluated against HF guidance and current design practices. The following two issues were identified and resolved.

Issue: RCS flow indication.

The onsite review indicated that the parameter RCS flow is not available on the DPS or DIAS. The CSF walkthroughs indicated that this is a valuable parameter. Requested ABB-CE to state its position regarding the inclusion of RCS flow indications in the DIAS.

Evaluation: ABB-CE stated that the Function and Task Analysis shows no use of RCS flow, RCP differential pressure, SG differential pressure, or core differential pressure to verify RCS loop circulation. Loop differential temperature, hot and cold leg temperatures, core exit thermocouples, and subcooling are required by Emergency Operations Guidelines (EOGs) to determine the existence of circulation. There is a low flow reactor trip on steam generator differential pressure to protect against a sheared shaft event. RCS flow is not required.

The absence of RCS flow on the DPS and DIAS was found acceptable after it was determined that RCS circulation could be acceptably inferred from existing plant parameters and instrumentation.

Issue: Display glare.

A potential problem with display glare was noted during the onsite review. There was a concern that the ambient lighting in the mockup room is dimmer that the anticipated CR conditions. In addition, the selection of display devices that are different from those used in the mockup may increase glare. Requested ABB-CE to describe the measures that will be taken to reduce glare in the final design.

Evaluation: The ABB-CE HFESGB document provides guidance that adequately addresses the issue of glare. The selection and installation of hardware for the final design will be performed in accordance with this document. ABB-CE stated that it will continue to evaluate display hardware for its acceptability in reducing display glare.

Display glare will be evaluated during suitability verification. ABB-CE's treatment of glare was found acceptable based on ABB-CE's commitment to design and review the HSI based on the guidance of the HFESGB document, which addresses glare.

18.6.1.3.1.5 Component Control System Process Controller Display

The onsite review examined the following CCS process controller displays: pressurizer pressure control resident on the RCS panel and pressurizer level control (charging and letdown) resident on the CVCS panel. The displays were presented on electroluminescent devices located on the bench-board section of the control panels. The following control modes were examined: master control level (e.g., RCS inventory), subloop level (e.g., charging), and component level (e.g., valve).

The CCS process controller display is an input device that combines the controllers for physically dissimilar but functionally-related systems into a single device that permits manual or automatic control at a number of different hierarchical levels (master, subloop, and component). This concept generally supports operator monitoring and control activities by organizing controls and displays by functional relationships and task requirements rather than by the physical relationships of plant equipment.

18.6.1.3.1.5.1 Review of the CCS Process Controller Display - Design Characteristics

An initial review of the 9 characteristics (see Section 18.7.1 of CESSAR-DC,) associated with the CCS process controller display found the following characteristics to be acceptable: 1, 3, 5, 6, 7, and 8. Characterisitics 2, 4, and 9 required additional review.

Characteristic 6 addresses the HF concern that in traditional CRs operators must adjust and monitor numerous controllers that together achieve a higher-level control goal. For example, a set of separate, hardwired controls may be physically separated on the control panel by the type of plant system that is to be controlled (e.g., fluid system controllers may be separated from electrical system controllers). The CCS process controller display combines control interfaces for a variety of plant component/systems into a single interface. For example, controls for pressurizer heaters and spray are combined into a single controller. Characteristic 7 addresses the fact that the CCS process controller provides the operator with three levels of control: master control, subloop control, and component control. Characteristic 9 addresses the fact operator inputs at these levels include selection of manual or automatic control modes, selection of control signals for loop control, and selection of loop control set points. This configuration reduces the operator workload associated with monitoring and adjusting a dispersed set of controls.

Characteristics 1 and 7 refer to the fact that the CCS process controller display provides software-generated representations of control devices and associated controlled variables. Control is accomplished by accessing the appropriate control device in the dynamic section of the CCS process controller display unit and then providing appropriate control inputs via touch-sensitive input devices. These characteristics, as represented by the current state of the design, are consistent with operator task requirements for accessing control and display devices, providing control inputs, and monitoring status. The location of the CCS process controller display (characteristic 3) maintains the separation of displays on the vertical section and input devices on the bench portion of the panel. Characteristic 5, which refers to the fact that the controller is read from the panel, is consistent with HF guidelines that state that character size should be consistent with the viewing distance of the operator's tasks. Since the controller is operated from the panel, this viewing distance is appropriate.

Characteristics 2, 4, and 9 received additional review because their implications to overall HSI and plant performance required further analysis.

Characteristic 2: Touch screen VDU devices are utilized.

Evaluation: In the CCS process controller display, touch screens are used to access process controllers and provide control inputs such as selecting manual or automatic control modes and adjusting controller set points. The following concerns were reviewed: repeated input actions may cause arm fatigue and the user's hand or arm may obscure some portions of the display. These issues were evaluated during the onsite review using the mockup and were not found to be problematic. Use of the touch screen is not anticipated to cause arm/hand fatigue due to the position of the CCS process controller display and the fact that this control is not anticipated to be used frequently. The user's hand/arm obscures very little of the display and the display is blocked only momentarily. Therefore, this characteristic was found to be acceptable.

Characteristic 4: DIAS dedicated parameter displays are configured to conform to the System 80+ human factors standards, guidelines, and bases.

Evaluation: The HFESGB document was reviewed separately and found to be acceptable as a basis for the HSI design.

Characteristic 9: CCS process controller is a man-machine interface only. All control loop electronics are located outside the MCR.

Evaluation: HICB was requested to review this characteristic and provided the following evaluation, which confirms the characteristic description. A further discussion of the CCS process controller and its acceptability from an I&C perspective is provided in Section 7.7. of this document.

ABB-CE states that the portions of the EFS and process component control systems that are located in the main control room and the remote shutdown panel are man-machine interfaces only. All control loop electronics are located outside the main control room. The staff concurs.

18.6.1.3.1.5.2 Review of the CCS Process Controller Display - Design Implementation

The CCS process controller display for pressurizer level control was examined and evaluated against HF guidance and current design practices. The following three issues were identified and resolved.

Issue 1: Deviation bar chart - salience of origin.

Evaluation: The zero value of this scale is not marked with a horizontal line to the right of the scale. This may hinder the operator's ability to rapidly assess conditions by requiring the operator to first determine which end of the bar is the origin. This is particularly important when the bar is short and both ends of the bar are near the origin.

ABB-CE stated that this is a prototype implementation issue that will be corrected by design review. The scale zero value will be marked to make it more salient. Hence, this criterion is satisfied.

Issue 2: Deviation bar chart - normal control band.

Evaluation: The normal operating range is indicated by a vertical band along the scale. This band is thin and not highly salient. This makes comparison of the bar to the normal range difficult. This is a problem with the pressurizer pressure control resident on the RCS panel. ABB-CE stated that the CCS process controllers will not be implemented using that display unit but instead with the same type of display unit as the pressurizer level control of the CVCS panel. The deviation bar charts of the pressurizer level controller were found to be acceptable. Hence, this criterion is satisfied.

Issue 3: Deviation bar chart - scale resolution.

Evaluation: The deviation bar charts for charging and letdown have scale demarcations in units of 10, with a range of -20 to +20, while the actual values are presented with a resolution of a single unit (e.g., 0.01m^3/min (3 gpm). This appears to conflict with guidelines that state that the resolution of the scale should match the resolution requirements of the user's information requirements. ABB-CE was requested to provide a justification for not providing a scale with a finer level of resolution.

ABB-CE stated that this is a prototype implementation problem that will be corrected in design review and that the scale resolution will match requirements for use. ABB-CE subsequently committed to modify Section 18.5.1.5.3 of CESSAR to state that parametric requirements for display and control variables will be defined in terms of precision in addition to the current dimensions of device type, range, accuracy, and units of measure. ABB-CE also committed to modify Section 6.1.5.2 - Phase 2 Availability Inspection Criteria of its verification and validation plan (NXP80-IC-VP790-03) to indicate that precision specifications will be verified for each as-built item of the HSI. This response was found acceptable because it provides a commitment to systematically evaluation display precision requirements and verify precision in the as-built display. Hence, this concern is resolved.

Issue 4: Accidental input or activation prevention.

Evaluation: The CCS process controller display does not provide safe guards for preventing accidental activation such as confirmation steps. This is in apparent conflict with guideline 3.1-4 of the NUREG/CR5908 which indicates that protection should be provided to prevent accidental actuation.

Evaluation: ABB-CE demonstrated actuation of the CCS process controller display using the mockup. Two features reduce the likelihood of accidental activation. First, direct access to controlled components is prevented by a summary display. For actuation to occur an input display, rather than the top-level summary display, must be present in the dynamic area of the CCS process control display. This feature is only effective if the previous user had returned the summary display to the dynamic area after using an input display. Second, accidental activation is protected by the way in which the touch screen operates. To activate a touch area the operator must move a finger into the touch target area to provide an input signal and then remove the finger without entering the inactive area around the poke area. This requires a fairly deliberate action. Simply sliding one's finger across the screen should not activate a touch area.

ABB-CE stated that if an accidental activation did occur, the consequences would be no worse than if the component actuation were due to a single failure or push button switch misoperation (e.g., loss of electrical power). Additionally, ABB-CE stated that when specifically required by fluid system design, provisions to preclude inadvertent component actuation are provided in the CCS (e.g., component breaker rackout, administrative controls).

ABB-CE stated that it is not its policy to provide confirmatory messages for the CCS process controller displays because these messages do not effectively prevent errors of intent and the consequences of erroneous input are not immediate and severe. The absence of confirmatory messages was found to be acceptable based on the justification provided by ABB-CE.

18.6.1.3.1.6 CCS Switch Configuration

The onsite review examined the CCS switch configuration for the CVCS makeup system on the CVCS panel. In addition, non-functioning mockups of the switches were observed at the RCS panel for control of reactor coolant pumps and backup pressurizer heaters. The switches consisted of physical push buttons with backlit legends that indicated operating status. Color coding was used to indicate status (e.g., red = active, green = inactive). In addition to status conditions, the component discrepancy state was examined. A component discrepancy occurs when the demanded state of a component (demanded by remote, automatic control action) is different from the actual state of the component. This condition was indicated by the flashing of the red or green backlit portion of the switch associated with the current state of the component.

The CCS switches are spatially-dedicated, functionally-grouped input devices that allow the operator to provide discrete control inputs (as opposed to control set points for automatic controllers as with the CCS process controller). This conceptual design generally supported operator requirements for controlling components and monitoring component status. The design and placement of the switches was found to be acceptable based on consistency with HFE guidelines for coding, spatial dedication and functional grouping and on functional similarity to other existing control panel switches. The acceptability of specific characteristics of the switches are reviewed in the sections below.

18.6.1.3.1.6.1 Review of the CCS Switch Configuration - Design Characteristics

The six characteristics (see Section 18.7.1 of CESSAR-DC,) associated with the CCS switch configuration were reviewed and found to be acceptable.

Characteristic 3: refers to the fact that CCS switch configuration devices are organized at control panels based on their functional relationships to the plant systems that are controlled from that panel. In addition, individual switch devices are grouped by functional relationships. At the RCS panel the switches were grouped in vertical columns by reactor coolant pump and control train. At the CVCS panel the switches were organized in a mimic configuration to reflect the control relationships between the individual switches and the overall plant system. This organization is consistent with HF guidelines related to functional organization of information to match operator tasks.

The CCS switch configuration uses physical push buttons with backlit legends (characteristic 1) that are consistent with traditional CR input devices. The ability to perform on-line replacement (characteristic 2) was successfully demonstrated and found acceptable.

Characteristic 4: refers to the location of the CCS switch configuration device on the bench board section of the control panels. This location was found acceptable; it maintains the separation between controls, which are located on the bench section, and displays, which are located on the vertical section. This configuration is consistent with HF guidelines that state that controls should be located such that their operation does not obstruct the operator's view of necessary displays.

Characteristic 5 states that CCS Switch Configurations conform to the System 80+ HFESGB. This document was subject to a separate review and was found to be acceptable as a basis for the System 80+ HSI.

Characteristic 6, which refers to the fact that the controller is read from the panel, is consistent with HF guidelines that state that character size should be consistent with the viewing distance of the operator's tasks. Since the switch is operated from the panel, this viewing distance is appropriate.

18.6.1.3.1.6.2 Review of CCS Switch Configuration - Design Implementation

The CCS switch configuration devices for the backup pressurizer heaters were examined and evaluated against

HF guidance and current design practices. The following issue was identified and resolved.

Issue 1: Accidental input or actuation prevention.

Guideline 3.1-4 of draft NUREG/CR-5908 states, "The system should prevent or minimize the accidental manipulation of control and input devices which could result in changes to the status of the system functions, components, or data."

Evaluation: During the onsite review, these switches were evaluated for resistance to accidental actuation. It was found that the raised bezel around the switch provides limited protection against accidental activation. Also, switch operation does not include confirmatory messages.

Guideline 3.1.5 of the HFESGB states that interface hardware should be designed and located so that accidental activation is unlikely. This guideline will be included in ABB-CE's verification of the final design. In addition, ABB-CE agreed to investigate methods to further reduce the probability of accidental component actuation. ABB-CE entered Item 79 into the TOI to record its commitment to address this issue. Hence, this criterion is satisfied.

18.6.1.3.1.7 Integrated Process Status Overview

As part of the walkthroughs, the onsite review examined the IPSO, which was implemented on a large rear-projection display screen at the front of the CR mockup. This overview of the plant included a critical function matrix, success path status indications, digital or trend indications of key parameters, and alarm presentations. The success path status indications used color-coded triangular symbols to indicate the operational status of individual success paths (collections of plant systems and equipment that are required for achieving specific CSFs). The symbology, placement, and color coding of the success path status indications were examined.

One HF concern in advanced CRs, where the majority of the displays are presented in "compact workstations" on largely computer-based display devices, is the potential inability of the crew to maintain a unified overall view of plant status. The IPSO addresses this concern by providing important plant status information to the entire crew and support staff (e.g., STA) in a fixed location, permanent display. The intention is to provide a common frame of reference and enhance the situation awareness of the individual crew members who are working at their own workstations and viewing different plant data.

The need for a spatially-dedicated, permanent display of plant status is the technical basis for the SPDS requirement and is generally recognized in current nuclear industry standards such as IEC 964, ALWR URD, and draft NUREG/CR-5908. In addition, support for the IPSO's achievement of this objective comes from a study conducted at the Halden Project, in Norway (HWR-184). The display was found to facilitate a rapid assessment of plant state and support operator detection and diagnosis of transients, although some problems were noted.

18.6.1.3.1.7.1 Review of the IPSO - Design Characteristics

Seven issues were identified and resolved.

Issue 1: Selection of plant parameters for the IPSO.

The selection of plant parameters for the IPSO is based on twelve critical functions, which are related to safety and power production. High-level alarm status boxes are provided on the IPSO display for these critical functions. Associated with each critical function are a number of plant parameter/indicators that are represented numerically (e.g., reactor power) and/or with symbols (e.g., atmospheric dump valve status).

The following parameters, identified in NUREG-1342 as important SPDS parameters, were either not present or not fully implemented on the IPSO mockup. These parameters are listed below according to the safety functions used by NUREG-1342

- Reactivity control
  - source range
- Reactor coolant system integrity
  - containment sump level
- Radioactivity control
  - effluent stack radiation
  - steamline radiation
  - Containment radiation
- Containment conditions
  - containment pressure
  - containment isolation status

Evaluation: In the April 19 through 21, 1993, meeting minutes, the NRC staff stated that this issue can be acceptably resolved by ABB-CE's commitment to incorporate these parameters, plus containment hydrogen concentration, into the IPSO display and to enter this commitment into its open issue tracking system. ABB-CE agreed to include these parameters and entered Item 62 into the TOI to record its commitment. Hence, this criterion is satisfied.

Issue 2: Trend indication symbols for key parameters.

Evaluation: The IPSO display includes arrows beside the digital values of key parameters to indicate the direction of change. This concept is consistent with Guideline 1.3.5-6, Direction of Change in Digital Display, of draft NUREG/CR-5908. Hence, this concern is resolved.

Issue 3: Logic for indicating the operational status of CSF success paths.

The status of plant systems such as safety injection (SI) is indicated on the IPSO with a triangle symbol. A solid green triangle indicates that the system is inactive. An open triangle with a red outline indicates that the system is active. The logic for indicating the operational status of success paths was found to be a potential source of confusion for operators. For example, the volume control system is not designated as active unless both the charging and letdown systems are active (i.e., if letdown is active and charging is inactive the volume control status is indicated as inactive). This may be misleading to operators because the net effect to the volume control system is to change (e.g., reduce) the RCS inventory. ABB-CE was requested to provide a rationale for the logic used to determine system status.

Evaluation: ABB-CE stated that the rule for system activity is that if any redundant train of a system is active, then the system is active. This a prototype implementation problem. The application of this rule for indicating the status of charging and letdown will be addressed by design review and prototype implementation.

This response was considered to be acceptable.

Issue 4: Measurement units for plant parameters.

Evaluation: The IPSO does not show units of measurement for temperature, pressure, level, and power for the depicted plant parameters. This is in conflict with HF design guidance including Guideline 1.3.6-6 of draft NUREG/CR-5908. ABB-CE stated that the HFESGB document will be updated to require that measurement units be included for all numerically displayed values. ABB-CE entered Item 80 into the TOI to record its commitment. The staff found this response acceptable based on the requirements to resolve all TOI issues as part of the V&V activities required by ITAAC and DAC.

Issue 5: System symbols.

Plant systems such as SI are indicated on the IPSO with a triangle symbol. Three guideline discrepancies regarding the use of this symbol were identified.

First, the triangle symbol is depicted in a vertical orientation for some systems, such as SI and in a sideways orientation for other systems, such as feedwater (FW). The sideways orientation is apparently intended to depict direction of flow. However, it is in conflict with Guideline 1.3.4-8, Upright Orientation, of the Guide, which states, "Icons and symbols should always be oriented upright."

Evaluation: ABB-CE entered Item 81 in the TOI to record a commitment to provide proper orientation for system triangle symbols. Hence, this issue is resolved.

Second, the triangle symbol has the same size, shape, and line width as the delta symbol used for RCS temperature. This is in conflict with Guideline 1.3.4-5, Distinguishability, of draft NUREG/CR-5908, which states, "Each icon or symbol should represent a single object or action, and should be easily discriminable from all other icons and symbols."

Evaluation: ABB-CE entered Item 82 in the TOI to record a commitment to revise the IPSO design so that the system symbol and delta symbol are different in size. Hence, this issue is resolved.

Third, the triangle symbol has the same line width and a similar shape as the symbol for a valve. While the symbols are different they do have significant similarities that may affect the operator's ability to rapidly comprehend the information presented by the display. This is also in conflict with Guideline 1.3.4-5, Distinguishability, of draft NUREG/CR-5908.

Evaluation: ABB-CE stated that there is sufficient difference between the valve symbol, which has been adopted from plant P&ID drawings, and a triangle. The triangle represents a new symbol for the operator to learn. No potential errors have been identified or are expected. V&V will confirm that this is not a problem.

This response was found satisfactory based on ABB-CE's commitment to confirm the adequacy of these symbols during verification and validation. Hence, this issue is resolved.

Issue 6: Symbols for direction of flow.

The triangle symbol used to represent plant systems is utilized to indicate direction of flow of major plant systems (e.g., the FW symbol points toward the steam generator to indicate FW flow). This is in conflict with Guideline 1.-2.8-5, Directional Arrowheads, of draft NUREG/CR-5908, which states, "Flow directions should be clearly indicated by distinctive arrowheads." The symbol for system should not be manipulated to serve the function of an arrowhead.

Evaluation: ABB-CE entered Item 81 in the TOI to record a commitment to provide proper orientation for system triangle symbols. ABB-CE stated that no arrowheads are needed to indicate direction. ABB-CE's response was found to be acceptable because the inclusion of the TOI item will ensure consistent use of the triangle symbol. The absence of arrow heads was found acceptable because the representations of plant systems were not sufficiently complex to require arrowheads to clarify the direction of flow.

Issue 7: Trend indication symbols for key parameters.

Evaluation: The IPSO display includes arrows beside the digital values of key parameters to indicate the direction of change. In addition, plus and minus symbols are used to indicate, respectively, that the current value is above and below the normal control bounds. The arrows are coded in white when the values are inside normal control bounds. The arrows and plus/minus symbols are coded in yellow when the values are outside of normal control bounds. The implementation of this coding scheme is consistent with the coding principles of consistency, simplicity, and discriminability described in Guideline 3.4.1-1 of draft NUREG/CR-6105. The use of color coding is redundant with the use of the plus and minus symbols and is therefore consistent with Guideline 1.3.8-11, Redundant Color Coding, of draft NUREG/CR-5908. Hence, this issue is resolved.

18.6.1.3.2 OER Item Review

As part of the overall design features review, items were selected from the OER in order to determine, on a sampling basis, if operating experience was appropriately considered in the design of the System 80+ CR. Three specific areas were selected:

- Generic Issue 23, RCP seals as a sample USI/GSI
- low-power and shutdown issues
- System 80 operating experience issues based upon interviews with System 80 operators

The following are the results of these reviews.

18.6.1.3.2.1 Generic Issue 23 - RCP Seal Related Instrumentation and Operator Response

For the review of Generic Issue 23, NUREG/CR-4544 - Reactor Coolant Pump Seal Related Instrumentation and Operator Response was reviewed to determine pertinent items that should have been considered for incorporation into the CR design. The primary concern was instrumentation and displays to prevent or anticipate possible seal failures and to cope with the occurrence of failures. Parameters identified as desirable were seal pressure, seal temperature, seal leakage, injection flow and temperature, seal staging flow and temperature, CCW flow and temperature, RCP shaft vibration related measurements, and overall RCS related parameters. All of the noted parameters were addressed by the ABB-CE design. Trending of parameters is also important. The DPS allows the operator to obtain trend plots for DPS parameters.

NUREG/CR-4544 also discusses the desirability of establishing diagnostic aids for trending seal degradation. The diagnosis of potential seal failures is difficult and such aids are pump vendor specific. Example charts are given in Figures A-3 and A-4 of that report. Charts such as these could be implemented quite well in a computer-based CR environment, where the parameters necessary for the chart (e.g., seal leakages and interseal pressures) are already monitored. ABB-CE has added Item 42 to their HF issues tracking system as a commitment to evaluate the need for this and other operator decision aids.

18.6.1.3.2.2 Low-Power and Shutdown

For the low-power and shutdown area, GL 88-17 and the ABB-CE shutdown risk report were reviewed. This review focused on two key concerns (1) two independent, continuous temperature indications representative of core exit conditions when the RCS is in a mid-loop condition, and (2) two independent, continuous RCS water level indications whenever the RCS is in a reduced inventory condition. A related concern regarding level indication is ensuring adequate overlap between the scales for normal level and the reduced inventory level.

The System 80+ design includes core exit thermocouple temperatures, hot leg temperatures and a new core heated junction thermocouple probe. These would appear to satisfy the temperature monitoring requirement. However, the details of these instruments and their displays are not yet designed. ABB-CE also included reactor vessel level, refueling level, and pressurizer level instruments in the design. Again, the HSI design of the reactor vessel level and the refueling level instruments and displays is not complete. ABB-CE stated that adequate overlap would be provided between the reactor vessel level and the refueling level instruments. Alarms related to these instruments are also not yet complete.

Therefore, ABB-CE has taken appropriate actions to address the low-power and shutdown area, given the current stage of the System 80+ design, and continues to give this important area attention. It is anticipated that those areas that have not yet been designed will receive similar attention by ABB-CE as the design process proceeds.

18.6.1.3.2.3 System 80 Experience

The third area examined for RCS-related OER items was the review of a number od System 80 operator interviews. These are summarized below, along with the status of the ABB-CE design in each area. Subsequent to the onsite reviews, ABB-CE provided a formal response to the System 80 operator interviews (Ref. 4 of CESSAR-DC Section 18.10, LD-93-135).

(1) Tracking of heat-up and cool-down rates: ABB-CE developed a proof-of-principle DPS screen to aid the operators in tracking these rates. Further work is needed and will be done during the design process. An example of one area needing improvement is the cool-down rate, which as currently provided, is only based on a one-hour time frame. In addition to this one-hour-based rate, the operators need a rate that is based on a much shorter time interval for control purposes. ABB-CE stated that consideration would be given to that during the design process.

(2) Operator decision aids to assist in initial post-trip actions: ABB-CE is considering such an aid but has not yet developed it. Item 41 has been entered into the TOI system to address this issue.

(3) Exploration of automation of RCP seal isolation: ABB-CE entered Item 86 in the TOI to record its commitment to evaluate the need for this and other operator decision aids.

(4) Operator decision aid for calculation of primary leak rate: ABB-CE stated (Ref. 4 of CESSAR-DC Section 18.10, LD-93-135) that they intend to develop such an aid during the detailed design process.

(5) Mid-loop reduced inventory operations: System requirements have been established; however, the HSI has not yet been designed. This should proceed well, as noted in the previous section, due to the attention given to shutdown by ABB-CE in their shutdown risk report.

(6) Pushbutton lamp replacement is problematic: ABB-CE adopted a design that will alleviate this problem by using a special tool for removing the light fixture, replacing the bulb in the fixture, and then reinserting the fixture. ABB-CE demonstrated the light bulb replacement task during the May 13 and 14, 1993, onsite review. With careful reinstallation of the fixture, inadvertent actuation of the push button can be avoided.

(7) Improved means to manually depressurize: As compared to the System 80 design, ABB-CE made considerable improvements in this area with the reactor depressurization system (RDS), the reactor coolant gas vent system, and integrated controls and displays for pressurizer control.

(8) Use of units familiar to operators: Only one issue was noted in this area, namely the lack of RCS flow measurement indicated in percent of full-power flow rate. This was later determined to be acceptable based on a review by NRC Reactor Systems Branch, which determined that such an indication was not required.

In summary, ABB-CE has appropriately addressed, for the current stage of the design, the incorporation of operating experience into the design of the System 80+. Further, ABB-CE has made a commitment to continue to evaluate operating experience (in the form of new industry and government reports and other applicable documents) and to incorporate issues, as identified, into the design. Specific items have been entered into ABB-CE's HF issue tracking system to ensure that they receive attention during the detailed design process.

18.6.1.3.3 DSER Issues

The following DSER Issues are relevant to the design features review

- 18.8.1 - Shape Coding Used to Prioritize Alarms
- 18.8.1.3 - Flash Coding of Alarms
- 18.8.1.4 - Size Coding of Alarms
- 18.8.2 - Additional HSI Information Required for Staff Review

DSER Issue 18.8.2 contains 19 sub-items (see Section 18.6 of this report under "DSER Review"). The following sub-items were relevant to this review:

(a) Provide human engineering justification for the

  (5) alarm scheme
  (6) interactive display hierarchy

(b) Provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for

  (1) IPSO
  (2) Alarm scheme and alarm acknowledgement
  (3) Display hierarchy and navigation scheme used for CRTs

18.6.1.3.3.1 DSER Issue 18.8.1 - Shape Coding Used to Prioritize Alarms

As a result of the meeting of September 10 and 11, 1992, ABB-CE committed to provide information to describe its rationale for shape and salience coding of alarms.

Evaluation: The alarm system employs a coding scheme to express two dimensions of alarm importance: priority and state. Alarm priority is based on the proximity of the alarm setpoint to a significant operator action condition. Alarms are organized into three levels of priority; priority 1 being the last warning prior to a significant operator action condition, priority 2 being the next to last warning, and priority 3 being any number of warnings prior to the next to last warning. The alarm priority scheme, including six categories of significant operator actions, was reviewed as part of the review of design methods and general characteristics and found to be acceptable. Alarm priority is represented by the following shape codes; priority 1 - an illuminated box, priority 2 - an illuminated frame, and priority 3 - illuminated brackets (four corners of the frame).

Alarm state has four levels (new, existing, cleared, and reset) that are coded by tile intensity and flash rate. These are applied to the shape (e.g., reverse video, frame, or brackets) surrounding the alarm tile. New alarms have the highest intensity shape and flash with a 50/50 on-off cycle. Existing alarms have an intermediate level of intensity and do not flash. Cleared alarms have the lowest level of brightness and flash with a 25/75 on-off cycle. Reset alarms have no illumination.

ABB-CE described a design process in which various design concepts for the alarm tiles were generated, subjectively evaluated, and modified. ABB-CE described informal experimentation and subjective evaluation of various alternatives for the alarm coding scheme. No formal process for collection and analysis of empirical data was presented by ABB-CE. While the information coding schemes are consistent with general HF guidance for information coding, concerns were identified regarding the specific coding values that were implemented in the design. ABB-CE entered Items 74, 75, 76, and 78 into the TOI system to record its commitment to address these concerns.

While individual dimensions of the alarm coding scheme may be consistent with HF guidelines, the effectiveness of the overall alarm coding scheme, including the integration of shape, flash, and intensity codes remains largely untested. ABB-CE entered Item 77 into the TOI to record its commitment to evaluate the effectiveness of the alarm system through verification and validation activities when the system is fully implemented. In addition, ABB-CE entered Item 101 into the TOI to record its commitment to evaluate the alarm system using a prototype of the DIAS alarm tile prior to verification and validation.

Based on these commitments from ABB-CE, it was recommended that DSER Issue 18.8.1.1 is resolved.

18.6.1.3.3.2 DSER Issue 18.8.1.3 - Flash Coding of Alarms

As a result of the meeting of September 10 and 11, 1992, ABB-CE committed to provide a rationale for the alarm flash duty cycle that is 50/50 on-off for new alarms and 25/75 on-off for cleared alarms. This rationale was to include a justification for inconsistency with NASA 3000, "NASA Man-Systems Integration Standards," (1989) which states "Flashing lights shall have approximately equal amounts of ON and OFF time."

Evaluation: The alarm system uses flashing as a coding scheme to draw attention to those changes in alarm states that require an acknowledgement from the operator (i.e., new and cleared alarms). Because multiple alarm conditions are associated with each alarm tile, the flash rate was configured to allow more than one alarm state to be conveyed. For example, the coding shape for a new alarm is visible during the ON portion of its duty cycle and the coding shape for an existing alarm may be visible during the OFF portion of the new alarm's duty cycle. ABB-CE stated that the alarm system employs different flash rates for new and cleared alarms to compensate for the possibility that the flash rates for new and existing alarms may drift and overlap. If overlap did occur the new alarm would not be masked by the cleared alarm; the new alarm would be visible because it has a longer ON cycle than the cleared alarm. ABB-CE further stated that flash coding is redundant with intensity coding and that similar duty cycles are used for new and cleared alarms in traditional CRs that have tile annunciators.

The rationale for the flash rates is satisfactory. However, because the effectiveness of the overall alarm coding scheme, including the integration of shape, flash, and brightness codes is largely untested, ABB-CE agreed to enter a number of specific issues into its tracking system for open HF issues. These issues are discussed in the review of Characteristic 12 in Section 18.6.1.3.1.2 of this report. The inclusion of these items will ensure that they will be addressed in later phases of the design and evaluation process including V&V. Based on ABB-CE's action to include these items in its tracking system, it is recommended that DSER Issue 18.8.1.3 is resolved.

18.6.1.3.3.3 DSER Issue 18.8.1.4 - Size Coding of Alarms

As a result of the meeting of September 10 and 11, 1992, ABB-CE committed to clarify ABB-CE's use of size coding of alarms and its relationship to color, shape, and flash rate.

Evaluation: ABB-CE clarified its use of size coding of alarm priority symbols and its relationship to color, shape, and flash rate at the April 19 through 21, 1993, public meeting. The design specifications for alarm tile symbols indicate that symbols for new alarms are slightly larger than those for existing and cleared alarms. However, alarm state is primarily coded by color and flash rate. Operators are not expected to make discriminations between alarm tiles based on tile size. Therefore, this characteristic may be disregarded for the purposes of this review. Based on this review, it was recommended that this DSER issue be closed. Hence, this issue is resolved.

18.6.1.3.3.4 DSER Issue 18.8.2.a - Human Engineering Justification

ABB-CE was requested to provide human engineering justification for the following

(5) the alarm scheme
(6) the interactive display hierarchy

(5) <u>Alarm Scheme</u>

Evaluation: The alarm scheme was addressed in Section 18.6.1.3.1.2 in the review of the DIAS alarm tile display. Seventeen characteristics were reviewed and generally found acceptable based on compliance with HF principles and guidelines. Specific issues were raised during the review regarding the alarm flash rate and the lack of prior experience and research from which predictions could be made regarding the effectiveness of the alarm coding scheme as an integrated part of a human-system interface. ABB-CE has entered these concerns in its tracking system for open HF issues (see characteristic 12 in Section 18.6.1.3.1.2). Hence, this issue is resolved.

(6) <u>Interactive Display Hierarchy</u>

Evaluation: The interactive display hierarchy was addressed in Section 18.6.1.3.1.1 in the review of the DPS display hierarchy. Eleven characteristics were reviewed and generally found acceptable based on compliance with HF principles and guidelines. A specific concern was raised during the review regarding arm fatigue resulting from use of the DPS touch screen interface. ABB-CE addressed this concern by entering an item in its tracking system for open HF issues to ensure that ABB-CE evaluates whether an alternative interface in addition to the touch screen design is necessary (see characteristic 3 in Section 18.6.1.3.1.1). Hence, this issue is resolved.

18.6.1.3.3.5 DSER Issue 18.8.2.b - System 80+ Specific Studies

Requested ABB-CE provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for

(1) IPSO
(2) alarm scheme and alarm acknowledgement
(3) Display hierarchy and navigation scheme used for CRTs

Evaluation: In Reference 7 of CESSAR-DC Section 18.10, LD-93-005, ABB-CE responded to this DSER issue by citing (1) Halden Reactor Project reports that supported the inclusion of the IPSO into the System 80+ HSI, and (2) suitability analyses that were conducted by ABB-CE to evaluate

- alarm scheme and alarm acknowledgement
- display hierarchy and navigation scheme used for CRTs

The Halden Reactor Project reports (Gertman 1986 and Reiersen 1987) evaluate the usefulness of a large overview display. However, these studies are not specific to the System 80+.

The suitability analyses were described in Part C, of NPX-TE790-01. The format of each evaluation was the same: statement of the issue, recommendation and resolution.

Evaluation was subjective; no criteria for acceptability were specifically stated. No attempts to define the thresholds of acceptable and unacceptable human performance were presented.

Design justifications provided by ABB-CE throughout the review process have shown a heavy reliance on HF guidelines, as presented in ABB-CE's HFESGB document, and subjective evaluation based on previous design experience. Efforts to evaluate the limits of human performance during the design process through System 80+ specific experiments or other analyses are very limited.

ABB-CE committed to address issues of human performance during its verification and validation efforts using mockups and simulators that represent the final, integrated HSI design. In addition, ABB-CE committed to evaluate during verification and validation those issues that were identified by reviewers during the HF review of the HSI. Based on these commitments, it is recommended that these DSER items are resolved.

18.6.1.3.4 Review of Design Features Against Safety Parameter Display System Criteria

ABB-CE addresses the SPDS concerns and criteria via an integrated design rather than a stand alone add-on system, as used at most current operating plants. The System 80+ design includes a large format IPSO, viewable from anywhere in the controlling workspace. The IPSO information screen is also available on any DPS CRT display (in the CR, TSC, EOF, and remote shutdown panel.) Further, the SPDS parameters have been integrated into the various other panels and screens.

The System 80+ design was reviewed against the NUREG-0737, Supplement 1 criteria, as well as the additional guidance provided in NUREG-1342. These criteria and the evaluation of the related ABB-CE features are described below.

(1) Should provide a <u>rapid and concise</u> display of critical plant variables to CR operators.

The concise display of critical plant parameters is provided by the IPSO information screen, which is presented on the IPSO wall-mounted projection display as well as the DPS CRTs. The IPSO information screen provides in a clear, concise manner the selected plant parameters. Every two seconds new IPSO data is acquired and the IPSO displays are updated, which provides the necessary rapid response. This information is acceptable and resolves the SPDS rapid and concise requirement of Supplement 1 to NUREG-0737.

(2) Should be located <u>convenient</u> to CR operators.

The IPSO information is provided as a dedicated single page display at the top of the DPS display hierarchy. It is accessible from any DPS CRT in the CR, TSC, EOF, or remote shutdown panel. The IPSO is also displayed on a large overview panel visible to all personnel in the control room. This information is acceptable and resolves the SPDS convenience requirement of Supplement 1 to NUREG-0737.

(3) Will <u>continuously</u> display plant safety status information.

The large overview IPSO panel is continuously "on" and visible to all operators. The contents of the IPSO display either contain the information variables or contain alarm boxes in the critical function monitoring (CFM) matrix that alert the operator to discrepancies and then directs the operator to the required information in the CFM displays of the DPS. Information is immediately accessible from the DPS via touch screens. This information is acceptable and resolves the SPDS continuous display requirement of Supplement 1 to NUREG-0737 because it meets the intent of NUREG-1342, "A Status Report Regarding Industry Implementation of SPDS," in that defines the content of SPDS displays to provide information which is sufficient to represent plant status but is not so large that meaningfulness and accessibility are negatively impacted.

(4) Should have a high degree of <u>reliability</u>.

This item has been reviewed and found acceptable by the Instrumentation and Controls Branch. See detailed discussion above under DSER Issue 20.2-17 in Section 18.3.3.2.5 regarding SPDS availability and reliability. The staff finds that the information provided by ABB-CE regarding SPDS reliability is acceptable and, therefore, this requirement of Supplement 1 to NUREG-0737 is resolved.

(5) Shall be suitably <u>isolated</u> from electrical or electronic interference with safety systems.

This item was reviewed and found acceptable by the Instrumentation and Controls Branch. Therefore, this requirement of Supplement 1 to NUREG-0737 is resolved.

(6) Shall be designed incorporating accepted <u>HFE</u> principles.

The IPSO and MCC were designed according to the ABB-CE's HFESGB document. Additionally, the panel design was reviewed using portions of draft NUREG/CR-5908. The design was found to be generally acceptable, although some issues were identified. Acceptable resolutions were subsequently achieved for these issues. The detailed results of that review of the MCC and IPSO are discussed in Section 18.6.1.3.1.7 above. Overall, the design was found acceptable, and therefore, this SPDS requirement of Supplement 1 to NUREG-0737 is resolved.

(7) Minimum information displayed shall be sufficient to determine plant safety status with respect to five safety functions.

Additional information on the five safety functions is provided in NUREG-1342, which was used for this review.

(i) For reactivity control, the SPDS should display power range, intermediate range and source range reactor power. The IPSO has provisions to display all of these. However, the source range information had not yet been added to the design. Further, the source range information will be calibrated in counts per minutes while the power/intermediate range will be in percent. This was potentially confusing in the mockup because the IPSO information screen did not display measurement units. However, ABB-CE has committed to display all measurement units.

(ii) For reactor core cooling and heat removal, the SPDS in ABB-CE PWRs should monitor RCS level, subcooling margin, temperatures (T_h, T_c, core exit), and RHR flow. The IPSO has both reactor vessel and pressurizer level, an alarmed subcooling monitor, T_h, T_c, core exit thermocouples, and a shutdown cooling success path indicator for when shutdown cooling is in operation, which will give an alert when a failure or loss of flow occurs. The shutdown cooling success path indication was not fully implemented.

(iii) For RCS integrity, the SPDS should monitor RCS pressure, T_c, containment sump level, and for the steam generator (SG) - pressure, level, and blowdown radiation. The IPSO contains RCS pressure, T_c, and SG level and blowdown radiation (the last via the critical function monitor). SG pressure is needed for a SG tube rupture event. ABB-CE, however, is using main steamline radiation (through the radiological emissions CFM block) as a surrogate for SG pressure. The IPSO does not have containment sump level, but ABB-CE has indicated that sump level may be included as part of one of the other CFM blocks, which are not yet designed.

(iv) For radioactivity control, the SPDS should monitor effluent stack monitors, steamline radiation, and containment radiation. ABB-CE stated that these would be contained on the radiological emissions CFM block, which is not yet designed.

(v) For containment conditions, the SPDS should monitor containment pressure and isolation status. The IPSO has containment pressure and a containment isolation CFM block. The CFM block, however, is not yet designed in detail, nor described in the CESSAR-DC.

In the April 19 through 21, 1993, public meeting minutes, the NRC staff stated that this issue can be acceptably resolved by ABB-CE's commitment to incorporate into the IPSO display the missing parameters, plus containment hydrogen concentration. ABB-CE agreed to this an entered Item 62 into the TOI to record its commitment. The staff finds that ABB-CE's information and commitments discussed are acceptable, and therefore, this SPDS requirement of Supplement 1 to NUREG-0737 is resolved.

(8) Procedures and operator training, addressing actions with and without SPDS, should be implemented.

Currently ABB-CE does not intend to develop detailed procedures. By letter dated September 23, 1992, ABB-CE indicated that the System 80+ SPDS is being developed in a complementary (parallel) fashion with the development of System 80+ emergency operations guidelines. Further, ABB-CE indicated that in developing System 80+ guidelines which involve use of SPDS information, provisions for operating with and without critical functions monitoring are being made.

The minutes of the May 13 and 14, 1993, public meeting (dated July 15, 1993) note ABB-CE's position that operator training relative to the SPDS will be a COL action item. This COL action item was found to be acceptable.

In summary, the staff finds that ABB-CE's responses and commitments regarding the eight SPDS requirements of Supplement 1 to NUREG-0737 are acceptable, and therefore, GSI Issue I.D.2 is resolved. Hence, this issue is resolved.

(9) The staff's position regarding an exemption from 10 CFR 50.34(f)(2)(iv) for a plant safety parameter display console:

The regulation 10 CFR 50.34(f)(2)(iv) requires that an application:

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded (I.D.-2).

Combustion Engineering, as part of the System 80+ SSAR, commits to meet the intent of this requirement. However, as discussed below, the functions of the safety parameter display system (SPDS) will be integrated into the control room design rather than on a separate "console." The purpose of the requirement for an SPDS, as stated in NUREG-0737, Supplement 1, is to "... provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. ... and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core."

The System 80+ design does not provide a separate SPDS, but rather, the functions of the SPDS are integrated into the overall control room display capabilities. In lieu of the requirements in 10 CFR 50.34(f)(-2)(iv) for a "console," CE has proposed the following commitments in the System 80+ SSAR:

(1) Section 18.7.1.8.1, Safety-Related Data, states that the Nuplex 80+ Advanced Control Complex provides a concise display of critical function and success path performance indications to control room operators via the Data Processing System

(2) Section 18.7.1.8.1 states that the IPSO big board display is a dedicated display which continuously shows all critical function alarms and key critical function and success path parameters

(3) Section 18.7.1.8.1 describes the SPDS for the System 80+ and states that all five of the safety function elements are included in the DPS Critical Function Hierarchy which forms the basis of the Nuplex 80+ SPDS function:

(a) Reactivity control
(b) Reactor core cooling and heat removal from the primary system
(c) Reactor coolant system integrity
(d) Radioactivity control
(e) Containment conditions

(4) Section 18.7.1.8.2 states that the critical function and success path monitoring application in conjunction with the continuous IPSO display and the DPS CRTs meet SPDS requirements for Nuplex 80+ without using stand-alone monitoring and display systems

The Commission may, upon its own initiative or at the request of an applicant, grant exemptions from the requirements of the regulations of Part 50. The exemption must comply with 10 CFR 50.12 (a) criteria regarding special circumstances. An exemption from the "console" of the SPDS may be granted since not having an SPDS "console" (1) does not present an undue risk to the public health and safety, and is consistent with the common defense and security (10 CFR 50.12 (a)(1)); and (2) special circumstances exist that application of the regulation to the System 80+ design of the SPDS rule is not necessary to achieve the underlying purpose of the SPDS rule (10 CFR 50.12 (a)(2)(ii)). As presented here, the staff uses the special circumstances in 10 CFR 50.12 (a)(2)(ii) to justify the deviation from the regulation (exemption) for an SPDS "console" for the System 80+ design.

In conclusion, the staff finds an exemption from the requirement for an SPDS "console" to be appropriate based upon (1) the description in the CE SSAR of the intent of the System 80+ design to incorporate the SPDS function as part of the plant status summary information which is continuously displayed on the fixed-position displays on the large display panel; and (2) a separate "console" is not necessary to achieve the underlying purpose of the SPDS rule which is to display to operators a minimum set of parameters defining the safety status of the plant. The staff therefore finds that CE has adequately supported an exemption from 10 CFR 50.34(f)(2)(iv) because SSAR sections 18.2(6), 18.4.2.1(14), 18.4.2.8 and 18.4.2.11 achieve the underlying purpose of the rule by ensuring that the SPDS functional requirements are satisfactorily incorporated in the control room design without a separate "console."

18.6.1.4 Standard Design Features Findings

The seven design features addressed by this review were found to be generally consistent with HFE design principles and guidelines. Further, the HSI design appeared to adequately address SPDS criteria. In some cases specific concerns were identified that could not be resolved at this stage of the HSI design. ABB-CE has recorded these issues in its HF issue tracking system and has committed to address these issues in later stages of the design and evaluation process.

Design justifications provided by ABB-CE throughout the design features review have shown a heavy reliance on HF guidelines (as presented in ABB-CE's HFESGB document) and subjective evaluation. The review has shown a high degree of consistency between the design and these guidelines. However, some concerns regarding human performance are not directly addressed by available guidelines. Efforts by ABB-CE to evaluate issues of human performance during the development of HSI via System 80+ specific experiments or other analyses that measure human performance have been very limited. ABB-CE committed to address issues of human performance related to its final, integrated HSI design during its verification and validation effort. These evaluations will use mockups and simulators. This effort is addressed by HFE PRM Element 8 - Verification and Validation. In addition, ABB-CE committed to evaluate issues related to the alarm system using a prototype of the DIAS alarm system prior to verification and validation. Based on the review of ABB-CE's design features and its commitments to address human performance issues in later stages of the design process, the design features issues addressed by this report are resolved.

18.6.2 Human-system Interface Design Methods and General Characteristics

This section addresses the following

- the methods for implementing the display and control requirements, selecting hardware and software, and refining of design concepts
- design criteria used to determine CR and control panel arrangements including the overall configuration of the main control console and the position of individual control/display devices within individual panels
- general design characteristics that were incorporated into the HSI

The application of the methods and criteria to the design of the CR configuration, RCS panel, and remote shutdown panel is discussed. Relevant DSER issues are also addressed.

18.6.2.1 Objectives

The objective of this review is to ensure that the design methods and criteria used to determine the content and arrangement of displays and controls in the System 80+ are consistent with accepted HF guidance and practices.

18.6.2.2 Methodology

18.6.2.2.1 Description of Review Methodology

This review included a desktop review of design documentation as well as an onsite review of ABB-CE's System 80+ mockup. During the desktop review, ABB-CE design documentation was examined and evaluated against accepted HF design guidance and practices. During the onsite review, which was conducted as part of the System 80+ design features review, a selected set of design features were analyzed using walkthrough evaluations of operator tasks and evaluations using HF guidelines. This provided a practical format for examining the appropriateness of the design criteria and the acceptability of preliminary design products.

18.6.2.2.2 Material Reviewed

The following ABB-CE documents were referenced in this review:

- Reference 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 1 (untitled), attached response to RAI No. 620.2, ABB-CE letter dated May 8, 1992.
- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX80-IC-DP-790--01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.
- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), attached response to DSER Item 20.2-29," ABB-CE letter dated December 18, 1992.
- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18 DSER Open Item Responses," ABB-CE letter dated January 18, 1993.
- Reference 12 of CESSAR-DC Section 18.10, LD-93-1-35, "System 80+ Information for Issue Closure," Attachment 6, Sub-Attachment 1, "Comments from Draft TER (July 14, 1993) on Nuplex 80+ HSI Justification of ABB-CE Positions Requested for Closure of HSI Issues," ABB-CE letter dated September 1, 1993.

- Reference 13 of CESSAR-DC Section 18.10, LD-93-147, "System 80+ Information for Issue Closure," Attachment 1, "Response to Cross-Branch Chapter 19 Questions (October 4, 1991)," ABB-CE letter dated October 18, 1993.

- Reference 6 of CESSAR-DC Section 18.4, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+," (NPX80-IC-DR-791-02, Rev. 00), September 15, 1993).

- Reference 4 of CESSAR-DC Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design," (NPX80-IC-DP790-01, Rev. 02, September 29, 1993).

- CESSAR-DC, Sections 18.6, 18.7 and 18.8.

18.6.2.2.3 Design Criteria Documents

The following materials were consulted as part of this evaluation:

- Electric Power Research Institute (1990). Advanced Light Water Reactor Utility Requirements Document, Volume II, ALWR Evolutionary Plant, Chapter 10: Man-Machine Interface Systems.

- Human Factors and Ergonomics Society (1988). "American National Standard for Human Factors Engineering of Visual Display Terminal Workstations" (ANSI HFS-100), Santa Monica, CA: Human Factors and Ergonomics Society.

- IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE Standard 279, 1971.

- Kinkade, R.G. & Anderson, J. (1984). Human Factors Guide for Nuclear Power Plant Control Room Development (EPRI NP-3659). Palo Alto, CA: Electric Power Research Institute.

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), attached response to DSER Item 20.2-29," ABB-CE letter dated December 18, 1992.

- U.S. Code of Federal Regulations, Part 50.54 (k) and (m) U.S. Government Printing Office, Washington, D.C.

- U.S. Nuclear Regulatory Commission, (May 1973) "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," (RG 1.47), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1981). "Guidelines for Control Room Design Reviews" (NUREG-0700), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1989), "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems" (NUREG-1342), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1994), "Advanced Human-System Interface Design Review Guideline," (Draft NUREG/CR-5908), Washington, D.C.

- U.S. Nuclear Regulatory Commission (1989), "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit," (RG 1.114, Rev. 2), Washington, D.C.

18.6.2.2.4 Scope and Limitations

The focus of this review is on the design of the HSI both within and outside of the MCR. Within the MCR the review addresses the overall CR configuration and the design of panels at the MCC and auxiliary console and safety console (ACSC). Many of the details of the System 80+ CR design are not specified because the design is not yet complete. However, design details are provided by ABB-CE for most of the RCS panel of the MCC as an example of how the design methods and criteria will be applied by ABB-CE. This review also addresses HSI elements located outside the MCR including the remote shutdown panel and local control stations. The designs for the remote shutdown panel and local control stations were also not complete.

18.6.2.3 Results

The following is a review of the System 80+ HSI design methods and general characteristics based on relevant DSER issues and HFE PRM criteria. Unless specifically noted otherwise, ABB-CE responses to these issues and criteria are from LD-93-135, Attachment 6, Sub-Attachment 1: Comments from Draft TER (July 14, 1993) on System 80+ HSI Justification of ABB-CE Positions Requested for Closure of HSI Issues.

18.6.2.3.1 DSER Review

DSER Issue 18.8.2 contains 19 sub-items (see Section 18.6 above). The following sub-items were relevant to this review.

(a) Provide human engineering justification for

(1) control panel profiles
(2) control panel arrangement in the control room
(3) the selection of control devices
(4) the selection of the display devices

(b) Provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for the following sub-items:

(8) Auditable documentation of the design process that supports the human performance aspects of the reduction in the quantity of data presented to the operator

(9) Impact of human performance of the difference between breadth of information in System 80 and System 80+ CRs

(10) Qualitative and quantitative criteria that identify when the operator is receiving "enough" rather than "too many" or "too few" number of alarms and displays

(11) Auditable documentation to track the data/information that was lost/gained between System 80 and System 80+ CR designs

(12) Effects (positive and negative) on operators performance of the changes, individually and collectively, between System 80 and System 80+

These DSER items are addressed in Section 18.6.2.3.2. Table 18.5 provides a cross-reference between the DSER items and the subsections of this document where items are addressed.

18.6.2.3.2 HFE PRM Criteria-Based Evaluation

This section provides a review of the System 80+ based on the HFE PRM Element 6 criteria. Because of the broad scope of the HSI, this review has been divided into the following sections that correspond to major portions of the System 80+ HSI design:

- General evaluation - Issues that are relevant to the overall HSI review scope.

- CR configuration - Issues related to the overall configuration of the MCR including design methodologies, analyses, and products.

- Information presentation - Issues related to the design criteria and methods used for the depiction of plant information through controls and displays.

- Panel layout - Issues related to the design criteria and methods used for the organization of controls and displays within panels.

Also reviewed were

- reactor coolant system panel
- remote shutdown panel

These were provided by ABB-CE as examples of application of the HSI design approach.

18.6.2.3.2.1 General Evaluation

This section addresses evaluation issues that apply to the overall HSI design methods and criteria.

(1) Criterion: HFE PRM General Criterion 1 states that the design configuration shall satisfy the functional and technical design requirements and insure that the HSI will meet the appropriate HFE guidance and criteria.

Evaluation: Specific design issues related to functional and technical design requirements and HFE guidance and criteria are described in the design issues subsections of Sections 18.6.2.3.2.2 through 18.6.2.3.2.7. This HFE PRM criterion was held open until all specific design issues were resolved. Hence, this issue is resolved.

(2) Criterion: HFE PRM General Criterion 2 states that the HFE effort shall be applied to HSI both inside and outside of the CR (local HSI).

Evaluation: ABB-CE has committed to the overall scope in the HFPP and the staff has found the scope acceptable (the staff's review of HFPP scope is provided in FSER Section 18.2.3.2.1, "General Purpose, Scope, and Organization."

(3) Criterion: HFE PRM General Criterion 5 states that the HSI shall be free of elements which are not required for the accomplishment of any task.

Evaluation: The information/panel layout method described in Section 18.7.2 of CESSAR-DC includes reviews of information requirements to identify HSI elements that are necessary for operator tasks. This process requires that HSI elements have an established need before they are included in the HSI design and thus excludes unneeded HSI elements. Further, the V&V analyses are intended to

Table 18.5 DSER issue items

DSER Item	Section
18.8.2	
a.1	18.6.2.3.2.2
a.2	18.6.2.3.2.4
a.3	18.6.2.3.2.4
a.4	18.6.2.3.2.4
b.8	18.6.2.3.2.1
b.9	18.6.2.3.2.1
b.10	18.6.2.3.2.1
b.11	18.6.2.3.2.1
b.12	18.6.2.3.2.1

evaluate the availability of required displays and controls and the overall effectiveness of the integrated HSI for task performance. Hence, this issue is resolved.

(4) Criterion: HFE PRM General Criterion 8 states that the HFE/HSI problems shall be resolved using studies, experiments, and laboratory tests. Examples are:

- Mockups and models may be used to resolve access, workspace and related HFE problems and incorporating these solutions into system design.
- Dynamic simulation and HSI prototypes shall be evaluated for use to evaluate design details of equipment requiring critical human performance.

The rationale for selection of design and evaluation tools shall be documented.

Evaluation: Sub-Items a, b.8 and b.9 address the application of this criterion to:

- CR configuration
- Information presentation: symbols, formats, and other means
- Panel configuration
- Remote shutdown panel and the local control panels

(a) CR configuration.

Evaluation: Visibility and personnel mobility issues related to the design and arrangement of consoles in the control room were evaluated in CESSAR-DC Section 18.6.5.6. Specific concerns related to visibility between locations in the CR are reviewed in Issues 3 and 4 of Section 18.6.2.3-.2.2 and were found acceptable. This criterion is satisfied based on the analyses performed by ABB-CE.

(b) Information presentation: symbols, formats, and other means.

Evaluation: ABB-CE's discussion of symbols, formats, and other information presentation means provided in Section 18.7.1 of CESSAR-DC does not provide descriptions of empirical studies or other analyses that were conducted to resolve problems related to symbols, formats, and other information presentation means. ABB-CE's discussions of its design evaluation activities, found in LD-93-005, indicated that ABB-CE relied largely on HF guidelines rather than empirical studies or other tests. ABB-CE stated that the use of guidelines is acceptable and preferred as cost-effective, unless specific problems are identified which require a more resource-intensive approach for resolution. ABB-CE agreed to conduct additional testing of the DIAS alarm tile display system to address concerns regarding the coding of alarm information. This commitment is recorded in Item 101 of ABB-CE's HF TOI. This testing is to be conducted using a DIAS alarm tile display prototype prior to the verification and validation of the final HSI design. While human factors guidelines can provide an acceptable basis for HSI design, additional evaluations and tests should be performed to address design concerns that are not adequately resolved through human factors guidelines. This criterion is satisfied based on ABB-CE's commitment to conduct tests of the DIAS alarm tile display system to address specific design concerns.

(c) Panel configuration.

Evaluation: Requested ABB-CE to describe any studies that it conducted that addressed the resolution of panel layout design problems associated with the earlier stages of design. ABB-CE stated that its use of mockups remains ongoing and its use of task, availability, and suitability analyses culminate with the complete design but receive iterative efforts throughout the design process. This criterion was satisfied based on ABB-CE's iterative analyses that include use of the control panel mockup.

(d) Remote shutdown panel and the local control panels.

Evaluation: In CESSAR-DC Section 18.8, ABB-CE indicated that the remote shutdown panel has the same profile of the MCC in the MCR and is based on the criteria of that panel. However, ABB-CE has indicated that the designs for the remote shutdown panel and the local control panels are not complete. The remote shutdown panel and the aspects of the local control stations that are relevant to emergency procedures will be addressed through suitability and availability analyses and finally by validation testing. Additional analyses have not been proposed by ABB-CE. No design issues were identified that extended beyond the scope of those analyses that were already planned. Hence, this issue is resolved.

(5) Criterion: HFE PRM General Criterion 10 states that the HSI design elements shall be evaluated to assure their acceptability for task performance and HFE criteria, standards, and guidelines. Further, DSER Issue 18.8.2.b stated that ABB-CE should provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for:

- Auditable documentation of the design process that supports the human performance aspects of the reduction in the quantity of data presented to the operator
- Impact of human performance of the difference between breadth of information in System 80 and System 80+ CRs
- Qualitative and quantitative criteria that identify when the operator is receiving "enough" rather than "too many" or "too few" number of alarms and displays
- Auditable documentation to track the data/information that was lost/gained between System 80 and System 80+ CR designs
- Effects (positive and negative) on operators performance of the changes, individually and collectively, between System 80 and System 80+.

Evaluation: The five sub-items of the DSER are addressed below. The numbers of the sub-items correspond to those used in Section 18.6.

(a) Sub-Item b.8 - Auditable documentation of the design process.

Evaluation: ABB-CE's response in LD-93-005 describes the measures taken to reduce the quantity of data presented to the operator. This issue is resolved based on this approach for documenting HSI information and control requirements based on analyses of operator tasks and excluding information and controls that are not supported by a specified need.

(b) Sub-Item b.9 - Impact of human performance of the difference between breadth of information in System 80 and System 80+ control rooms.

Evaluation: In LD-93-005, ABB-CE states that there is no difference between breadth of information in the System 80 and System 80+ CRs, but there are differences in the way in which information is presented through the HSI. Based on a review of functional similarities conducted as part of HFE PRM Elements 3 and 4, there does not appear to be major differences in the breadth of required information. Availability analyses and integrated validation conducted as part of V&V will evaluate the adequacy of the breadth of information provided in the completed design. Hence, this issue is resolved.

(c) Sub-Item b.10 - Qualitative and quantitative criteria that identify when the operator is receiving "enough" rather than "too many" or "too few" number of alarms and displays.

Evaluation: The FTA is intended to define the minimal required set of information. In LD-93-005, ABB-CE indicated that this issue will be further evaluated during V&V, using the final design. Hence, this issue is resolved.

(d) Sub-Item b.11 - Auditable documentation to track the data/information that was lost/gained between System 80 and System 80+ control room designs.

Evaluation: In LD-93-005, ABB-CE stated that there is no requirement to track this difference, but that the effectiveness of the HSI will be evaluated during verification and validation. The function analysis reviewed in Section 18.4 identified differences in functions between the System 80 and System 80+ plants and determined that those functions of the System 80+ that were allocated to the operator were acceptable. HSI requirements will be derived from analyses of operator tasks associated with these functions. This issue was resolved because an acceptable approach is used for deriving HSI information and control requirements and because differences in the operator role which result from differences in the plants are identified and evaluated.

(e) Sub-Item b.12 - Effects (positive and negative) on operators performance of the changes, individually and collectively, between System 80 and System 80+.

Evaluation: In LD-93-005, ABB-CE stated that a direct comparison is not possible due to the lack of an adequate baseline. However, the effectiveness of the System 80+ design will be evaluated separately from the System 80 during verification and validation. This response was found to be acceptable. Hence, this issue is resolved.

(6) Criterion: HFE PRM General Criterion 11 states that design and evaluation efforts associated with the HSI shall be performed using a listed set of documents as guidance.

Evaluation: ABB-CE references HF sources in "Human Factors Engineering Standards, Guidelines, and Bases for Nuplex 80+," NPX80-IC-DR-791-02 (HFESGB). This issue was addressed in a separate review of the HFESGB document and found to be acceptable. Hence, this issue is resolved.

18.6.2.3.2.2 Control Room Configuration Design

Section 18.6 of CESSAR-DC describes the process by which the System 80+ CR configuration was designed including the arrangement of the MCC, the auxiliary console and safety console, and other control room features. The following issues were identified.

(1) Criterion: HFE PRM General Criterion 6 states that the selection and design of HSI hardware and software approaches shall be based upon demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options.

Evaluation: The CR configuration process described in Section 18.6 of CESSAR-DC is acceptable. This is based on reviews of the design criteria as well as the description of the process by which alternative designs were developed and evaluated. Specific design issues related to the CR configuration are presented below and addressed separately. Hence, this issue is resolved.

(2) Criterion: HFE PRM General Criterion 7 states that the HFE standards shall be employed in HSI selection and design. Human engineering guidance regarding the design particulars shall be developed by the HSI designer to (1) insure that the human-system interfaces are designed to currently accepted HFE guidelines and (2) insure proper consideration of human capabilities and limitations in the developing system. This guidance shall be derived from sources such as expert judgement, design guidelines, and standards, and quantitative (e.g., anthropometric) and qualitative (e.g., relative effectiveness of differing types of displays for different conditions) data. Procedures shall be employed to ensure HSI adherence with standards.

Evaluation: The application of HFE standards to the design of the CR configuration is reviewed below. Also addressed is DSER Issue 18.8.2, which requires "human engineering justification for control panel profiles." (See DSER Issue 18.8.2, Sub-Item a.1 in Sections 18.6 and 18.6.2.3.1 of this report.)

The design criteria for CR configuration described in Section 18.6.3 of CESSAR-DC include NUREG-0700 criteria for anthropometrics, line-of-sight to information and controls, and desk and chair design. These criteria were found to be acceptable. The process used to develop the System 80+ CR configuration was an iterative process that started with a Nuplex 80 control room as a baseline. Design alternatives were evaluated against operational and HF concerns. This process was found to be acceptable. However, the specific dimensions for the MCC and ACSC profiles are not provided in sufficient detail to allow the application of the process and design criteria to be verified. ABB-CE was requested to describe the control panel profiles in sufficient detail to demonstrate conformance to HF criteria. Control panel dimensions and relevant reach and vision envelopes were provided in Figures 18.6.5-11 and 18.6.5-12 of CESSAR-DC. These were reviewed in greater detail as part of the review of the HFESGB and found to be acceptable. Based on this review it was determined that both HFE PRM General Criterion 7 and the DSER Issue 18.8.2 that pertains to human engineering justification for control panel profiles are satisfied because acceptable design guidelines and quantitative data were used to provide an acceptable human engineering justification for the control panel profiles.

Design Issues Related to Control Room Configuration Design

The following issues are relevant to Criterion 1 as well as other criteria of HFE PRM Element 6. ABB-CE's responses to these issues are discussed. The full text of ABB-CE's responses may be found in Reference 14 of CESSAR-DC Section 18.10, LD-93-135.

(1) Issue: Section 18.6.6.1 of CESSAR-DC states a minimum CR ventilation rate of 0.42 cubic m (15 cubic ft) of air per minute. NUREG-0700 states a minimum of 0.42 cubic m (15 cubic ft) per minute per room occupant. Since a supervisor may be frequently present in the CR, in addition to two operators, the minimum ventilation value stated by ABB-CE appears to be too low. ABB-CE was requested to address this apparent discrepancy.

Evaluation: In its response, ABB-CE stated that Guideline 7.3.1 of HFESGB correctly cites the NUREG-0700 guideline regarding ventilation and that Section 18.6.6.1 of CESSAR-DC was modified to indicate that control room ventilation rates will meet HFESGB guidelines. Hence, this issue is resolved.

(2) Issue: Section 18.6.6.1 of CESSAR-DC states that background noise levels will be in accordance with HFESGB, which states a maximum background noise level of 65 db(A) and a reverberation time of one second or less. This section also states that workstation lighting will be in accordance with HFESGB, which states detailed illumination criteria. ABB-CE was requested to specify how the environmental conditions of the CR will be evaluated. Will they be verified by ABB-CE in an ABB-CE facility or will this be a COL responsibility? What tools and methodology will be used for this evaluation.

Evaluation: ABB-CE stated that workspace (e.g., CR) environmental conditions will be evaluated through survey and measurement of the actual as-built facilities per applicable criteria from the HFESGB. This is part of the suitability inspection specified in the HF V&V plan, which in turn is part of the verification of suitability required by the HFPP and ITAAC items for the MCR and remote shutdown panel. Verification of environmental conditions is thus a COL applicant responsibility. Selection of personnel to perform the activity will be at COL applicant discretion. ABB-CE further stated that while the measurement (i.e., acceptance) criteria need to be specified, it is not necessary to specify measurement tools at this time. This position was found acceptable because necessary measurement tools can be more appropriately determined after the control room has been completed.

(3) Issue: Section 18.6.5.6.1.2 of CESSAR-DC states that the MCC is visible from a central location at either the auxiliary console (AC) or safety console (SC). However, from Figure 18.6.5-8, it appears that only a portion of the plant monitoring and control panel, and none of the RCS and CVCS panels are visible from the AC and SC. ABB-CE was requested to clarify its statement including a discussion of the possible effects of impaired visibility of the MCC. ABB-CE was also asked to address the apparent discrepancy between these visibility limitations and the design requirement for large digital readouts on the DIAS displays.

Evaluation: ABB-CE stated that visibility of the MCC area from the AC and SC panels is provided to facilitate operator communication and coordination, not for direct monitoring or reading activities, and is therefore acceptable. In addition, the CRT displays on the AC and SC panels provide access to all information available at the MCC. The IPSO provides plant overview information that can be read throughout the CR. DIAS digital displays are designed to be read across the MCC (e.g., read RCS panel DIAS displays while standing at the turbine panel), not across the CR. ABB-CE revised CESSAR-DC Section 18.6.5.6.1 to clarify the statements that refer to visibility. The visibility characteristic were found to be acceptable because they were consistent with tasks performed in the CR. ABB-CE's commitment to clarify the description in CESSAR-DC satisfies this concern.

(4) Issue: Section 18.6.5.6.1.4 of CESSAR-DC states that unobstructed visual access exists to the MCC from the CR supervisor (CRS) and shift supervisor (SS) offices. However, from Figure 18.6.5-9, it appears that the RCS panel is not visible from the shift supervisor's office. ABB-CE was requested to address this apparent contradiction.

Evaluation: ABB-CE stated that visibility of the MCC area from the CRS and SS offices is provided for general observation. It is not intended to support direct personnel supervision or plant monitoring. ABB-CE revised CESSAR-DC Section 18.6.5.6.1 to clarify the terminology and related subordinate statements. (See also Issue 3 of Section 18.6.2.3.2.2.) The visibility characteristic were found to be acceptable because they were consistent with tasks performed in the CR. ABB-CE's commitment to clarify the description in CESSAR-DC satisfies this concern.

(5) Issue: Section 18.3.2 of the CESSAR-DC discusses CR staffing and the design bases for the CR configuration. Section 18.6.1 defines the various terms used in describing the control room configuration, such as "controlling workspace," and "control room." 10 CFR Part 50 uses the terms "at the controls" and "control room." RG 1.114 provides guidance in detail as to what is meant by, and necessary for, these areas. One example is the need for an unobstructed view of controls, displays and alarms "at the controls." ABB-CE has not used the same terms and has not provided a commitment to RG 1.114, thus making it unclear as to their commitment to the detailed guidance and requirements of the RG and 10 CFR Part 50. ABB-CE was requested to provide such a commitment or alternatively describe clearly their method to be used in place of the RG. ABB-CE was also requested to provide one of its CR figures that clearly demarcates the pertinent areas.

Evaluation: ABB-CE stated that the design of the Nuplex 80+ control room will accommodate the COL applicant's meeting of the requirements of 10 CFR 50.54(k) and (m), and RG 1.114. However, RG 1.114 presents behavioral and administrative requirements on COL applicant operators, rather than design requirements; thus it does not form the basis for a coherent commitment by ABB-CE. Compliance with these issues, as RG 1.114 states, are COL applicant responsibilities.

Nonetheless, the Nuplex 80+ philosophy and design are cognizant of and consistent with the general intent of RG 1.114, i.e., to keep undivided operator attention focused on the plant. The following additions to CESSAR-DC aim to reinforce this point and address reviewer concerns.

CESSAR-DC Section 18.3.2 has been modified to further address control room staffing. The Nuplex 80+ controlling workspace is equivalent to the "surveillance area" specified in RG 1.114; an operator attending to and responsible for performing operations on the controlling workspace panels is considered to be "at the controls." Related definitional statements will be added to the discussion in CESSAR-DC Section 18.3.2, and the controlling workspace will be shaded in Figure 18.6.5-3. The "control room vital area" discussed in RG 1.114 is equivalent in System 80+ to the area within the control room security boundary identified in CESSAR-DC Chapter 13 Appendix A, Sections 2.2.I and 2.2.J.

This response was found to be acceptable because it clarifies ABB-CE's terminology and commitment to Regulatory Guide 1.114.

(6) Issue: Section 18.6.3 of CESSAR-DC states as a design criterion for the CR configuration that adequate work surface (laydown space) is provided at, or near, controlling workspace consoles for procedures, etc. without interfering with display viewing or control manipulation. However, specific criteria in terms of location and size are not provided. ABB-CE was asked to describe the measures that will be taken to ensure adequate laydown space.

Evaluation: ABB-CE stated that because it was unable to identify specific acceptance criteria on procedure laydown space in the general literature, the regulatory guidance, or the System 80+ design that it was reluctant to develop its own criteria. ABB-CE stated that rolling bookcases, two controlling workspace desks, and a clear area on the plant monitoring and control panel all provide procedure laydown space options for operators at the control panel area. The CRS console and the large desk behind it also provide laydown space for documents. ABB-CE added Item 92 to its HF issues tracking system to ensure consideration of this issue in subsequent design and verification and validation activities. Hence, this issue is resolved.

18.6.2.3.2.3 Information Presentation

This section provides a review of the methods and criteria used in the presentation of information on indications and controls in the System 80+ CR. The full text of ABB-CE's responses to these issues may be found in Reference 14 of CESSAR-DC Section 18.10, LD-93-135.

(1) Criterion: HFE PRM General Criterion 6 states that the selection and design of HSI hardware and software approaches shall be based upon demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options.

Evaluation: Software approaches for presenting plant information through symbols and graphical formats on computer-generated displays rely on HF guidance embodied in ABB-CE's HFESGB document to which ABB-CE has committed to use as its HF standard. This document has undergone a separate review and was accepted. Hence, this issue is resolved.

(2) Criterion: HFE PRM General Criterion 7 states that procedures shall be employed to ensure HSI adherence with standards. ABB-CE was requested to describe measures that will be taken to ensure that information presentation conventions/criteria will be systematically applied during the development of the balance of the System 80+.

Evaluation: ABB-CE stated that the HFPP provides a formal structure for disseminating the HFESGB to the design team. The use of standard features and computer aided engineering (CAE) tools in the detailed design will provide configuration control over the implementation of design conventions. Suitability verification ensures that conventions and criteria have been acceptably applied. These mechanisms collectively form an integrated and comprehensive approach to this aspect of design control. In addition, ABB-CE has entered an item in its HF issue tracking system to ensure that the reference design for the MCR and remote shutdown panel indications and controls shall be detailed using a systematic process incorporating HFE guidance. Hence, this issue is resolved.

(3) Criterion: HFE PRM General Criterion 9 states that HFE shall be applied to the design of equipment and software for maintainability, testing, and inspection. Software maintenance/upgrade is a growing concern as a source of failure for complex human-machine systems. Software plays critical roles in many aspects of the System 80+ including sensor data processing and verification, alarm processing, and control/display aspects of the user interface. ABB-CE was requested to describe provisions for protecting against loss of software integrity due to maintenance/upgrade work that will be performed by ABB-CE and by the utility.

Evaluation: ABB-CE stated that the HFESGB provides some guidance applicable to software maintainability. Verification by memory checks passed to the DPS for comparison with stored values will serve to reduce entry errors. Development and maintenance of all software is governed by the System 80+ Software QA program (NPX80-SQP-0101.0). Hence, this issue is resolved.

Design Issues Related to Information Presentation

(1) Issue: Item G of Section 18.7.1.1.2 describes the use of the color orange to code "operator established information." However, it is unclear from the description which information is being referred to (e.g., operator established alarms? operator aids?). ABB-CE was requested to clarify this statement including the type of information that will be coded and methods by which this coding that will be applied.

Evaluation: ABB-CE stated that operator established alarms are yellow, consistent with all other alarms. Operator aids are orange to denote non-ordinary, non-alarming conditions. CESSAR-DC Section 18.7.1.1.2 has been revised for clarity. Hence, this issue is resolved.

(2) Issue: Section 18.7.1.1.2 states that in the case of loss of indication from a valve, the position prior to instrument failure is displayed and the instrument failure condition is indicated with an asterisk placed before the valve symbol. Since the actual valve position may be different from the position prior to instrument failure, the displayed position may be misleading. ABB-CE was requested to provide a rationale for presenting this position versus other options such as indicating the position as unknown.

Evaluation: ABB-CE stated that power for position indication and power for valve movement are typically provided by separate circuits and mechanisms. Loss of indication does not imply other changes in component status. Thus, the last indicated state remains informative as the best estimate of actual state; to not display it would be to discard information. This position was found an acceptable approach for presenting information to the operator in the case of a failure.

(3) Issue: In the case of fault select the operator may select sensor channels via the discrete (DIAS) monitor. How will the sensor channels be selected on the DPS for the same parameter? How will sensor selection be handled for parameters that are only displayed on the DPS and not on the DIAS?

Evaluation: ABB-CE stated the controlled-access keyboard interface identified in Item 4, below, will be used for the selection of sensors for the DPS displays. The keyboard interface is located at the CRS console. It was found acceptable because the CRS console is readily accessible from the MCC and access to sensor channel selection is already available at the MCC via the DIAS displays.

(4) Issue: Section 18.7.1.1.8 of CESSAR-DC indicates that component or parameter information unavailable from automated data acquisition means are entered manually into the DPS. ABB-CE was requested to describe the human-computer interface that will be used for entering data including data entry screens, methods of interaction, provisions that will be made to reduce input errors, and provisions that will be made to ensure that entered values are kept current.

Evaluation: ABB-CE stated that a controlled-access interface for data entry (i.e., a keyboard) has been functionally specified. The interface will be located on the CRS console and in MCR office(s). An item has been entered in ABB-CE's HF issue tracking system (Item 95) to ensure treatment of this human-computer interface in subsequent design and V&V activities. Hence, this issue is resolved.

(5) Issue: Section 18.7.1.1.4 of CESSAR-DC describes the assignment of alarms into categories (e.g., priorities 1 to 3 plus a fourth category called operator aids). This assignment is based on the proximity of the alarm setpoint to the significant operator action conditions. The following issues were identified.

(a) The meaning of the term "significant operator action" should be defined. This definition should include the implications for automatic system actuations. For example, if a condition will result in the activation of an automatic protection system (e.g., safety injection or reactor trip) no operator action may be required. In this case is the alarm considered to be high priority?

Evaluation: ABB-CE stated, "Significant Operator Actions are those judged to be necessary to prevent specific undesirable consequences; these will often be redundant with automatic (i.e., protective) actions (for defense in depth). Alarms are not associated with automatic actions, per se." This position was found acceptable because it emphasizes the use of alarms to alert the operator to the need to take action and makes a distinction between alarm messages and other messages that indicate changes in plant status such as the actuation of an automatic system.

(b) How are alarms that are relevant to multiple conditions addressed? For example, if a single alarm is the third warning before a CSF violation and the last warning before a success path availability violation, would the alarm be assigned priority 3 or 1? This discussion should discuss whether some significant operator action conditions are of more importance to operator action than others (e.g., CSF violation versus major damage to equipment).

Evaluation: ABB-CE stated, "The alarm priority scheme does not focus on prioritizing the relative importance of alarms because this is a context-dependent judgment that remains the operator's responsibility. Rather, the priority scheme seeks to provide a strict ordering of priorities within dimensions that can be aggregated as the level of abstraction increases. Where a condition leads to redundant alarms, the alarm judged to be of lesser importance will be incorporated or suppressed." This position was found acceptable because the ordering priorities indicates alarm importance while allowing the operator to determine the relative importance of alarms within the same category based on the context of the specific plant event. This aids the operator while keeping the operator involved in the decision-making process. Also, the suppression of lower priority alarms is an acceptable alarm processing method for reducing operator workload

(c) Section 18.7.1.1.4 of CESSAR-DC implies that all personnel hazard alarms are priority 1 since these alarms often have a single setpoint. ABB-CE was requested to clarify this section of CESSAR-DC.

Evaluation: ABB-CE stated that all radiation alarms are not priority 1. Smoke/hazard alarms will be prioritized using the same criteria as other alarms. Such personnel hazard alarms could easily be prioritized on the basis of exposure limits and toxicity. This may result in adding criteria to those already stated, as is practically necessary to impart meaningful organization to the alarm scheme. Any added or revised rules will be incorporated in the alarm system design documentation. This issue will continue to receive consideration as one of several being tracked on the alarm system. Hence, this issue is resolved.

(d) ABB-CE stated that additional rules may have to be added before the alarm categorization is complete. ABB-CE was requested to document these rules in their design documentation and to keep these records current.

Evaluation: ABB-CE stated that any added or revised rules will be incorporated in the alarm system design documentation. Hence, this issue is resolved.

(6) Issue: Section 18.7.1.1.7 of CESSAR-DC specifies that the IPSO shall be readable from the shift supervisor's office. ABB-CE was requested to identify the criterion that will be applied for legibility of the IPSO from the shift supervisor's office.

Evaluation: ABB-CE stated the criterion of 15 minutes of arc (MOA) for minimum character height found in Section 18.7.1.1.7 of CESSAR-DC was outdated and was revised to indicate a minimum height of 12 MOA for any specified reading distance. This will make CESSAR-DC consistent with HFESGB Section 2.2.3.2.b. Letter heights on the IPSO (5.3 cm (2.1 in.)) at the specified reading distance from the SS office (approximately 12m (40 ft)) yield a proximal character height of 15 MOA. Since these values are ultimately based on the position of the reader with respect to the display, and since there are no tasks outside the controlling workspace that preclude viewers from adjusting their position for a better view of IPSO, ABB-CE considered the present character sizes to be acceptable.

The acceptability of ABB-CE's legibility criterion of 12 MOA, rather than a value of 15 MOA as specified by NUREG-0700 or 16 MOA as specified by ANSI-HFS-100, was addressed by the HFESGB review. ABB-CE's rationale for using 12 MOA as a robust criterion for legibility was not found to be fully supported by HF literature. However, it was acknowledged that actual viewing distances may be less than design assumptions stated in HFESGB. ABB-CE agreed to include Item 102 in its HF issue tracking system to ensure that legibility be further evaluated under conditions that are representative of anticipated work conditions. The character height of 5.3 cm (2.1 in.) for the IPSO was found acceptable on the basis that it resulted in a visual angle of 15 MOA, not 12 MOA, when viewed from the SS office. Hence, this issue is resolved.

(7) Issue: Section 18.7.1.1.7 of CESSAR-DC states that a DPS CRT located on an adjacent panel may be used by the operator to support monitoring tasks. CESSAR-DC, Amendment E stated that data on CRTs is sized for readability assuming the largest CR panel size. In CESSAR-DC Amendment N, that sentence was omitted. ABB-CE was requested to describe the degree of legibility (e.g., only alarm symbols versus all text) required of CRTs on adjacent panels to support operator task requirements. ABB-CE was also requested to describe how the DPS screens will be designed (including design criteria for character heights) to address these viewing requirements.

Evaluation: ABB-CE stated that although DPS screens do not normally need to be read from adjacent panels, the specified DPS screen character size (4.4 mm (.175 in.)) yields a proximal character height of 12 MOA at the specified reading distance (127 cm (50 in.)). This value is sufficient between panel centers, and is reasonably robust to off-angle viewing (see basis for HFESGB Section 2.2.3.2.b). The 12 MOA value also meets the criterion of NUREG-0700 Section 6.7.2.2.b(1) for character size on CRT displays.

Verification of legibility of CRTs from adjacent panels is addressed by HF issue tracking Item 102. Hence, this issue is resolved.

(8) Issue: Section 18.7.1.6.2 of CESSAR-DC discusses various mechanisms for controlling and indicating components and systems. Items addressed include: control location, engineering safety features actuation system (ESFAS) control signals, bypassed and inoperable status, interlocks and actuation signals, etc. While many positive features were noted, the description of the operator override scheme for actuated signals contained in Sections 18.7.1.6.2.2 through 18.7.1.6.2.5 seemed contradictory and was not completely clear. The following are specific examples.

(a) Section 18.7.1.6.2.2 states that <u>no</u> alarms or status indication is required for override at the component level. However, per RG 1.47 and IEEE 279, an override at the component level should give a system-level bypass or override indication.

Evaluation: ABB-CE stated that overrides and bypasses are separate and distinct entities. Overrides are a component-level control capability for manual action following automatic action. Override may follow, but cannot prevent (or meaningfully precede) an automatic actuation. Override does not produce a unique indication or annunciator per se; the manually operated component simply indicates its new operating state (i.e., active/inactive). However, this does not rule out that the new component state may in turn cause one or more alarms of various types.

RG 1.47 requires automatic indication in the control room of bypassed/inoperable status at the system level . . . of the protection system and the systems actuated or controlled by the protection system. Execution of a component override could conceivably change a safety system's status to bypassed/inoperable, in which case indication would be required per RG 1.47. However, component override capabilities and safety system bypass/inoperable status indication requirements are independently determined in all cases.

Based on this explanation and a review of additional details regarding bypassed/inoperable status indication that were provided in CESSAR-DC this concern was considered to be resolved.

(b) Section 18.7.1.6.2.2 states that there is an operator override for all ESFAS signals, yet 18.7.1.6.2.5 states that interlock signals cannot be overridden by the operator.

Evaluation: ABB-CE stated that both are correct, because interlocks and overrides are not equivalent. Section 18.7.1.6.2.2 states that the operator can override the ESFAS signals on any individual ESF-actuated component (override is not a system-level capability.) However, interlocks cannot be overridden. An interlock inhibits specific control action (either manual or automatic) until the condition(s) monitored by the interlock's sensor(s) are satisfied. Note: Interlocks are provided for component/system protection and safety (e.g., the SIT isolation valve cannot be shut if the RCS pressure is above the SDC entry pressure). No interlocks have been identified that prevent automatic actuation of a safety system component.

This explanation was found acceptable,in that it clarifies the meaning of the terms so that there is no longer an apparent contradiction and the basic requirements of RG 1.47 and IEEE 279 are satisfied.

(c) Section 18.7.1.6.2.5 states that a component generally remains in the actuated state when the control signal clears, yet Section 18.7.1.6.2.2 indicates that the component will always remain in the pre-cleared state.

Evaluation: ABB-CE stated that the actuated state typically is the pre-cleared state. The actuated state is the position called for by ESFAS, and is the position a component will be in when the ESFAS signal clears unless (1) it was manually repositioned following initiation (i.e., overridden), or (2) it did not respond to the initiating automatic signal in the first place. ABB-CE also stated that there are currently no instances in the design where a component changes state upon the clearing of the control signal. If this feature is implemented, CE stated that each case would be evaluated and dispositioned during the detailed design process implementation. This explanation was found acceptable, because of the verification and validation requirements placed on the detailed design process implementation.

(d) Section 18.7.1.6.2.2 states that operator override capability is provided on all ESF actuated components, yet Section 18.7.1.6.2.5 states that in some cases an actuation signal can be overridden by the operator.

Evaluation: ABB-CE stated that ESFAS and actuation signals are defined separately. Operator override capability is generally provided for all ESFAS signals (although the design permits exceptions to this feature, none have been identified to date.) Actuation signals have three types. Priority 2 and 3 signals can be overridden; priority 1 signals (no override) are typically used for equipment protection (none identified for ESFAS signals to date). However, some interlocks will prevent the operator from overriding specific components if permissive conditions are not met (e.g., SIT isolation valves discussed in Issue 8b of Section 18.6.2.3.2.3.) The staff reviewed CE's response and prioritization scheme and found that it adequately addressed the staff's questions regarding operator override and ESFAS actuation. Further evaluation of the ESFAS system and specific interlocks is provided in Section 7.3 of this document.

(9) Issue: It was recommended that both the NRC I&C and the Reactor Systems Branches review Section 18.7.1.6.2 since there is material in it pertinent to their review areas.

Evaluation: ABB-CE entered Item 99 into its HF issue tracking system and provided acceptable responses to the results of these reviews in Reference 13 of CESSAR-DC Section 18.10, LD-93-147. The issues identified by the staff and responses provided by CE in LD-93-147 focussed on ensuring consistency in instrumentation descriptions and listing (e.g., nomenclature used, ranges for indications) within CESSAR. CE's responses indicated that various sections of CESSAR would be modified to ensure consistency. The staff reviewed the responses in LD-93-147 and the associated CESSAR modifications and found that CE had adequately addressed the staff's concerns. This issue is resolved.

(10) Issue: ABB-CE stated that the status of unavailable equipment will be explicitly provided through DPS and the success path monitoring system. ABB-CE was requested to describe specifically how the following conditions will be indicated to the operator via these two systems

- bypass and inoperable status
- tagout status (how controls will be "tagged")
- blocked status (how controls will be physically blocked)

In addition, ABB-CE was requested to describe how the blocking and tagging functionality that exists in traditional plants will be provided in the System 80+. The response should include instrumented and non-instrumented components and their representation on the DPS, CCS process controllers and CCS switches. For example, will tagout information be presented in the message area of the DPS after a symbol is poked? Will this representation be different on the CCS process controller? How will this information be conveyed for the CCS switches? How will equipment be blocked (isolated) from operation. How will this be presented to the operator? Will the representation be changed manually or automatically? Will physical barriers be provided such as covers over controls?

Evaluation: ABB-CE stated that unavailable equipment will be presented using the separate code conventions for alarms (yellow rectangles, etc.) and uncontrollable equipment (cross-hatching). Instrumented and not-instrumented components will, where their status is indicated, apply similar conventions. Status explanations will be provided on the DPS via point-poke messaging in the standard message area. Switches and discrete components will possibly continue to use tags; use of [physical] covers is not necessary or anticipated. The equivalent treatment for process controllers has not yet been determined. Input of not-instrumented component status information is intrinsically an administrative control issue. On the other hand, how control blocking will be implemented in the CCS is a design issue and has yet to be addressed. ABB-CE acknowledges this to be an important set of issues; their detailed treatment has been entered as TOI Item 96 for future treatment. Based on the evaluation and disposition of all TOI issues as required in the detailed HFPP, this issue is resolved.

(11) Issue: A review of the design documents and the HSI mockup indicates the apparent lack of a systematic approach for determining the degree of precision with which data are presented, whether in digital or graphic form, to the operator via the HSI. Section 18.5.1.5.3 of CESSAR-DC states that parametric requirements for display and control variables will be defined in terms of device type, range, accuracy, and units as part of the FTA methodology. While accuracy of data is an important requirement, it is a separate concern from the precision with which data are presented to the operator (e.g., the number of significant digits in digital displays, the number of intervals on scale displays). Specific examples of the lack of clearly defined display precision requirements were observed during the design features review with respect to the scaling on bar charts and other indicators. The ABB-CE HFESGB document provides general criteria for scaling but this is insufficient for determining the precision requirements for specific parameters. ABB-CE was requested to define a systematic process by which precision requirements will be defined for displayed values.

Evaluation: ABB-CE agreed to enhance the guidance for specifying precision that is contained in the HFESGB Section 2.4.3. ABB-CE modified Section 18.5.1.5.3 - Information and Control Requirements of CESSAR-DC to require that the precision requirements for each measured variable be specified based on an analysis of operator task requirements. In addition, ABB-CE modified Section 6.1.5.2 of the V&V plan to ensure that precision requirements are verified for each as-built control or display item. This is acceptable because it assures that display precision is defined and verified.

(12) Issue: The DIAS alarm tile display system assigns sets of alarm states to individual alarm tiles. The use of alarm list displays may become cumbersome or ineffective if an excessive number of alarm states are assigned to individual tiles or if the total number of alarms states assigned to an alarm tile display device is excessive. ABB-CE was requested to describe design criteria for the maximum number of alarm states associated with (1) a single DIAS alarm tile, and (2) a single DIAS alarm tile display device.

Evaluation: ABB-CE stated that there are no firm human performance criteria limiting the number of alarms within one tile or display device. ABB-CE generally acknowledges the concern for excessive alarms within a tile as a possible downside on the revised depth/breadth tradeoff of conventional control rooms, but believes that larger problems of conventional alarm systems (e.g., excessive breadth) have been mitigated, achieving a net usability improvement. In addition, the ratio of the expected number of alarm variables (1500) to active alarm tiles (400) is only about 4-to-1, a quite manageable average figure. Ultimately, the breadth of alarm activity must be faced by operators somewhere in any design (e.g., alarm logs); the question is whether the burden is being reasonably managed.

ABB-CE subsequently agreed to evaluate this issue further using a prototype of the DIAS alarm tile display system prior to verification and validation of the final design. This commitment is recorded in Item 101 of ABB-CE's HF issue tracking system. Hence, this issue is resolved.

18.6.2.3.2.4 Panel Layout

This section addresses issues related to the design criteria and methods used in the arrangement of controls and displays within panels and the arrangement of panels in the CR as presented in Section 18.7.2 of CESSAR-DC.

(1) Criterion: HFE PRM General Criterion 3 states that the HSI design shall utilize the results of the task analysis and the I&C inventory to assure the adequacy of the HSI.

Evaluation: The design method described in Section 18.7.2 of CESSAR-DC described the process by which operator functions are organized on the panels using the results of the FTA. Then controls and displays are organized within these functional groups using the results of the FTA and the I&C inventory. This requirement is satisfied because the design method described in Section 18.7.2 of CESSAR-DC utilizes the results of the task analysis and the I&C inventory along with human factors criteria and methods to assure the adequacy of the HSI.

(2) Criterion: HFE PRM General Criterion 6 states that the selection and design of HSI hardware and software approaches shall be based upon demon-

strated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options. Further, DSER Issue 18.8.2 states that ABB-CE should provide human engineering justification for the selection of control devices and display devices. (See DSER Issue 18.8.2, Sub-Items a.3 and a.4, in Sections 18.6 and 18.6.2.3.1 of this report.)

Evaluation: In LD-93-005, ABB-CE states that because switches and CRTs are extensively used in industry, HF was not extensively involved in their selection. ABB-CE's responses in LD-92-033 and LD-92-065 regarding RAI 62-0.2 indicate that the selection of flat panel hardware was primarily based on considerations other than HF. However, a preliminary evaluation was conducted to determine whether flat panel devices could provide required display features. Although human factors evaluations were not extensively involved in the selection of these HSI interfaces, ABB-CE did conduct human factors evaluations to evaluate the suitability of these interfaces to operator tasks. This criterion is therefore satisfied.

(3) Criterion: HFE PRM General Criterion 7 states that the HFE standards shall be employed in HSI selection and design. Human engineering guidance regarding the design particulars shall be developed by the HSI designer to (1) ensure that the human-system interfaces are designed to currently accepted HFE guidelines and (2) ensure proper consideration of human capabilities and limitations in the developing system. This guidance shall be derived from sources such as expert judgement, design guidelines and standards, and quantitative (e.g., anthropometric) and qualitative (e.g., relative effectiveness of differing types of displays for different conditions) data. Procedures shall be employed to ensure HSI adherence with standards.

Evaluation: The application of HFE standards to control panel layout is reviewed below. Also addressed is DSER Issue 18.8.2, which required "human engineering justification for control panel arrangement in the control room." (See DSER Issue 18.8.2, Sub-Item a.2 in Sections 18.6 and 18.6.2.3.1 of this report.)

The panel layout procedure described in Section 18.7 of CESSAR-DC is generally well organized and documented. It consists of three steps (1) determination of functional groups and assignment to respective control panels, (2) determination of required control and indication devices and assignment to appropriate functional groups, and (3) criteria and procedure for the detailed layout of controls and indications within functional groups. The methods for organizing HSI elements within functional groups include arrangement by flow path, sequence, and related function. Based on these considerations the panel layout approach was found to be acceptable. Control panel arrangements of the final design will be further evaluated during the verification and validation analyses. Hence, this issue is resolved.

Design Issues Related to Panel Layout

(1) Issue: CESSAR-DC Section 18.7.2.1.1.1 states that the MCC design basis requires that all controls and indications be provided to perform the following tasks:

   (a) Perform monitoring and control tasks associated with maneuvering the plant from hot shutdown to full-power operation and return to hot shutdown.

   (b) Monitor major automatic controls (i.e., pressurizer automatic pressure and level controls) to maintain plant availability.

   (c) Perform standard post-trip actions following a reactor trip.

   (d) Maintain monitoring capability of plant investment concerns.

It was noted that these criteria do not include:

- Perform plant heatup, cooldown, cold shutdown, and refueling.
- Perform monitoring and control tasks associated with normal plant operations.
- Perform monitoring to diagnose plant failures.
- Perform monitoring of CSFs and success paths to assess threats to plant safety and plan/select appropriate response paths.

  The importance of the MCC for responding to emergency conditions was demonstrated through prior analyses including the walkthrough evaluations conducted by the staff using emergency operations guidelines and CSFs. ABB-CE was requested to describe why the above criteria were not considered in the layout of the MCC.

Evaluation: ABB-CE stated that the purpose of the cited design basis statements in CESSAR-DC is to generally define the role of the MCC as distinct from other controlling workspace facilities, which aims to minimize unnecessary movement during normal (i.e., frequent) operations. However, it should be evident from other portions of the review that the cited concerns have been and are being addressed by other aspects of the System 80+ design and design process, and are incorporated in the MCC (e.g., via DPS and IPSO).

The task analysis scenarios defined in Section 18.5.1.5.1 of the CESSAR-DC address the operator activities described above including those performed at the MCC. The task analyses will establish control and display requirements for the MCC. Therefore, these operator activities will be addressed by the MCC design process, although they are not explicitly stated as design criteria. Therefore this criterion is satisfied.

(2) Issue: CESSAR-DC Section 18.7.2.3 states that the criteria for the maximum height above the bottom edge of the upper panel for devices requiring operator touch are 51 cm (20 in.) and 81 cm (32 in.), respectively, for sit-down and standing panels. The derivation of these values, which is based on a fifth percentile female, is unclear. ABB-CE was requested to describe the derivation of the specific dimension used in the panel design.

Evaluation: ABB-CE stated that these specifications were removed from CESSAR-DC. Instead, Section 7.6 of HFESGB is referenced. The control panel vision and reach envelops described in CESSAR-DC Section 18.6 and the corresponding guidance found in HFESGB were reviewed as part of the HFESGB review found in Section 18.6.3 and found to be acceptable. Therefore this concern is satisfied.

(3) Issue: The six-step procedure for Part II Determination of Required Control and Indication Devices and Assignment to Appropriate Functional Groups described in Section 18.7.2.1.6 of CESSAR-DC (Amendment N) contains confusing references to previous text. For example, Step 5 of Section 18.7.2.1.6 refers the reader back to the same section (Section 18.7.2.1.6). ABB-CE was requested to review this section and make appropriate modifications.

Evaluation: ABB-CE has revised CESSAR-DC Section 18.7.2.1.6 (Amendment Q) to correct confusing references. Hence, this issue is resolved.

18.6.2.3.2.5 Application of the Design Method to Reactor Coolant System Panel

The design of the RCS panel is provided in Section 18.7.3 of the CESSAR-DC as a demonstration of the System 80+ standard design features, information presentation conventions, and panel layout method. The approach is to be applied to the other panels of the MCC and ACSC. The following issues were identified.

(1) Issue: In Sections 18.7.3.1 and 18.7.3.2 of the CESSAR-DC, it is stated that procedures were reviewed to determine if other functions or parameters are required for the RCS panel. ABB-CE was requested to describe which procedures were reviewed and how they were used.

Evaluation: ABB-CE stated that Palo Verde Nuclear Generating Station Procedures (Normal, Abnormal, Emergency, and Alarm Response) and ABB-CEN-152, Rev. 3 (Emergency Operations Guidelines) were reviewed. These were used to help identify functional groups, indications, controls, alarms, and system details that had not yet been specified in System 80+ documents.

Since Palo Verde is a System 80 plant and is the predecessor plant for the System 80+ design, this use of the Palo Verde procedures is appropriate. The particular Palo Verde procedures selected were also deemed appropriate. This together with the other elements of the Human factors Engineering Program Plan should serve to allow correct determination of functions and parameters. This acceptably addresses this issue.

(2) Issue: Section 18.7.3.2.1.3 of CESSAR-DC describes the DIAS dedicated parameter display for RCS hot-leg temperature. Figure 18.7.3-13 shows this display in the menu mode, which allows the operator to select the sensors that are used as input to the displayed value (e.g., during fault select conditions). After the operator has selected specific sensors and returned the display to the analog/trend mode, will the display show a full trend (e.g., 30-min) for the new sensor selection, or will the trend history begin with the time of selection? Does the DPS possess the capability to immediately generate a trend or will the trend start plotting at the time that the selection is made?

Evaluation: ABB-CE stated that a new process representation value can only be selected by the operator for the analog/trend display if a validation fault occurs. If new sensors are used to drive the process representation, the trend will continue adding new values to follow the old values (i.e., the trend will not restart, but continue). The DPS possesses additional capability to display historical data and initiate operator defined trends.

This response was found to be acceptable on the basis that the operator's access to parameter trends is supported in the case of sensor failures. (See Issues 7 and 8 of Section 18.6.2.3.2.5 for review of related concerns.) Hence, this issue is resolved.

(3) Issue: Figure 18.7.3-13 of CESSAR-DC indicates that the calculated values for cold-leg temperature for Loops 1 and 2 are on separate display pages of the DIAS display. What capabilities are provided to allow the operator to view the Loop 1 and 2 values together and facilitate comparison of loop values?

Evaluation: ABB-CE stated that the capability to view Loop 1 and 2 T_{cold} values together is provided on any (DPS) CRT via selected display pages, and could be provided on DIAS subpages. There is no task that requires such capability to be used routinely.

This issue was found to be acceptable based on the capability to perform cross-checks manually or by viewing adjacent DPS CRTs. (See Issue 6 of this section for review of a related issue.) Hence, this issue is resolved.

(4) Issue: The menu display for the acoustic leak monitoring system shown in Figure 18.7.3-18 is not consistent with other pages of this display in that it does not provide unique system identifiers. For example, the identifier Z-107 is not present for the RC-200 relief valve.

Evaluation: ABB-CE stated that the unique system identifier for the acoustic leak monitoring system in Figure 18.7.3-18 was omitted. These figures in SSAR-DC were provided as examples. The final design will have identifiers for all displays. Hence, this issue is resolved.

(5) Issue: It was recommended that the selection of RCS parameters to be displayed in the DPS and DIAS systems, including the selection of those to be displayed on dedicated displays, be reviewed by other branches of the NRC for concurrence.

Evaluation: ABB-CE stated that the minimum inventory of fixed location main control room (MCR) alarms, controls, and indications needed to complete tasks identified in the emergency operations guidelines (EOGs) and PRA (Probabilistic Risk Assessment) analyses are being reviewed by NRC branches other than Human Factors (e.g., Containment, Reactor Systems, and I&C). Additional parameters are defined based upon US NRC Regulatory Guidelines (e.g., RG 1.97), task analyses and requirements of the System designers and confirmed in Availability Verification. TOI entry 99 commits to consider the results of these reviews.

The subject NRC branches have reviewed the MCR minimum inventory and found it acceptable. A detailed discussion on the minimum inventory review is provided in Section 18.9.3.3 of this document.

(6) Issue: The DIAS multiple parameter display for the RCS, described in Section 18.7.3.2.1.3 of CESSAR-DC, contains 32 sensor or validation outputs that can be displayed one at a time. How would operators perform cross-checks between these values, which are displayed separately (e.g., compare pump differential pressures between reactor coolant pumps 1A, 1B, 2A, and 2B)? What provisions are made to facilitate cross-checks between values in the multiple parameter display, especially in the case of a failure of the DPS system? (It was noted that the DPS screens do not show pump differential pressure for all four pumps on the same screen, but provide separate screens for Loops 1 and 2).

Evaluation: ABB-CE stated that many parameters in the DPS are presented to facilitate comparisons within a single screen. Adjacent DPS screens further allow comparisons of multiple parameters with diverse screen locations (the Plant Monitoring and Control Panel provides two CRTs side-by-side, as shown in Figure 18.7.4-2. The DPS is a highly reliable system with a mean time to repair of 4 hours. There is no requirement during that time to do cross-check tasks; therefore, it is not justified to explicitly design for it. The historical data storage and retrieval (HDSR) system provides some cross-checking capabilities, but it is not presently known how many parameters may be displayed at once. DIAS provides display of all values on subpages, permitting cross-checks to be performed manually. This would be similar to performing cross-checks in current plants using control board meters. Hence, this issue is resolved.

(7) Issue: Section 18.7.3.2.1.4 of CESSAR-DC states that historical and trend data are available for <u>only selected</u> reactor coolant pump (RCP) parameters via the DPS. During the onsite review, ABB-CE stated that the ability to select <u>any</u> of the DPS plant parameters and generate a trend with the desired scale resolution will be provided via DPS. ABB-CE was requested to describe the trend capability to be provided via DPS and resolve the apparent contradiction regarding which parameters will be compatible with this capability.

Evaluation: ABB-CE stated that any DPS data point can be trended from the present time however, the HSI needs to be designed for this interface. The DPS HDSR function stores 750 analog data points (parameters) for historical trending at resolutions of 5 seconds and 10 minutes. TOI database entry 91 has been made to ensure that the HDSR HSI is detailed, and that parameters to be stored and the display resolution are defined. Hence, this issue is resolved.

(8) Issue: Section 18.7.3.2.1.5 of CESSAR-DC states that historical and trend data are available via DPS for all RCP seal/bleed system parameters. ABB-CE was requested to describe what it means by the terms "historical data" and "trend data" including a discussion of the time limits that are used for storing these data.

Evaluation: ABB-CE stated that historical data is archival data; it must be retrieved prior to display. It can be displayed in a time series format. Trend data is a time history plot of the most recent data over a specified short duration (e.g., 30 min). The purpose of a trend display is to explicitly present a timeserial view of the parameter's recent and ongoing changes. This supports extraction of higher order information (i.e., first and second derivatives), observation of process characteristics, and the extrapolation and prediction of future process values. Trend data is retained during display, not retrieved prior to display. TOI database entry 91 has been made to ensure that the HSI for the HDSR is provided and designed in accordance with HFPP requirements, and that the HDSR parameters to be stored and the data/display resolutions are defined. Hence, this issue is resolved.

(9) Issue: Sections 18.7.3.2.3.4 and 18.7.1.5.2 of CESSAR-DC describe priority 2 operator established alarms. Two concerns exist: alarm establishment and alarm presentation. Section 18.7.1.5.-6.D briefly describes the process by which the operator may establish new alarms, which includes accessing a database and entering new alarm setpoints. ABB-CE was requested to describe the interface to be used to perform this task including displays to be accessed and input devices used to supply setpoints and applicable alarm messages. With respect to the representation of operator established alarms, ABB-CE was requested to describe measures that will be taken to ensure that operator established alarms are not confused with each other or with standard plant-generated alarms. In addition, ABB-CE was requested to describe constraints on the number of parameters and the number of setpoints per parameter for operator established alarms and how operator established alarms will be managed across shift turnovers.

Evaluation: ABB-CE stated that the operator-established alarms have a dedicated alarm tile on each panel and each operator established alarm has a separate alarm message. However, the design details of the interface for operator established alarms are not yet completed. Item 87 has been entered into ABB-CE's HF issue tracking system to ensure that the identified concerns are addressed. Hence, this issue is resolved.

(10) Issue: Section 18.7.3.2.3.5 of CESSAR-DC states that priority 3 alarms are only available on the DPS and individual alarm tiles are not required for these conditions. Other sections of CESSAR-DC and the onsite review have demonstrated the use of priority 3 alarms on the DIAS alarm tile display. ABB-CE was requested to clarify this apparent contradiction.

Evaluation: ABB-CE stated that priority 3 parameters that do not degrade to priority 2 or 1 conditions are processed and displayed only by the DPS. DPS performs processing and display of all alarms and operator aids. CESSAR-DC Section 18.7.3.2.3.5 will be clarified with regards to the DIAS alarm system. The final RCS panel design was modified to incorporate the System 80+ standard features and conventions described in other sections of CESSAR-DC. Hence, this issue is resolved.

(11) Issue: Section 18.7.3.6 of CESSAR-DC states that an operator aid alarm tile is provided in the lower--right corner of the DIAS alarm tile display. Several issues are described below.

(a) Figures 18.7.3-39 apparently identifies this as a tile for operator established alarms, not an operator aid. Section 18.7.1.5.5 states that operator aids are only presented on the DPS. Is this tile actually an operator established alarm tile?

Evaluation: ABB-CE stated that this tile is actually an operator established alarm tile. Operator aids are only presented on the CRTs, as stated in CESSAR-DC Section 18.7.3.2.3.6. Hence, this issue is resolved.

(b) How many operator established alarm tiles will be provided per DIAS alarm tile display and per panel of the MCC?

Evaluation: ABB-CE stated that only one operator established alarm tile per panel will be provided. CESSAR-DC Section 18.7.3.2.3.4 was revised to clarify this. It was determined that one operator established alarm tile per panel was not likely to greatly increase operator workload. Therefore this concern was satisfied.

(c) How many plant parameters may be associated with a single operator established alarm tile?

Evaluation: ABB-CE stated that this is a design detail and is addressed by Item 87 of its HF issue tracking system as described in Issue 9 of this section. Hence, this issue is resolved.

(d) How may setpoints may be associated with a single parameter of an operator-established alarm tile?

Evaluation: ABB-CE stated that this is a design detail and is addressed by Item 87 of its HF issue tracking system as described in Issue 9 of this section. Hence, this issue is resolved.

(12) Issue: Section 18.7.1.5.5 of CESSAR-DC describes operator aids, as information that is helpful to the operator for plant control, but lower in priority than priority 3 alarms. Operator aid information will be presented on the DPS CRTs using an ". . .orange underline of the text of the information it applies to. The operator aid information flashes when unacknowledged and then may be acknowledged by the operator; however, there is no reset state." The following concerns were identified.

(a) The content and appearance of the operator aid should be described in greater detail. For example, where will the text reside (e.g, in the message window?, in the main part of the screen?).

Evaluation: ABB-CE stated that the operator aid text will reside in the message window on the lower part of the CRT screen. ABB-CE has entered Item 100 into its HF issue tracking system to ensure that it will provides an operator aid illustration in the future following further implementation of operator aids in the prototype. Hence, this issue is resolved.

(b) The coding scheme, which was an orange underline, appears to conflict with Section 18.7.1.1.2 of CESSAR-DC which states that the color white will be used for operator aids and orange will be used for operator established (alarm) information.

Evaluation: ABB-CE stated that an orange underline is used for operator aids. ABB-CE also stated that CESSAR-DC Section 18.7.1.1.2.G was corrected to say "operator aids" instead of "operator established information." Hence, this issue is resolved.

(13) Issue: Apparent inconsistencies were noted within the DPS with respect to abbreviations. For example, the DPS display, "Inventory Control (CFM) Level 2," shown in Figure 18.7.1-6 of CESSAR-DC provides a poke area labeled "PZR PRES" for quick access to a supporting diagnostic page. However, the corresponding designator of the PRI menu page shown in the Figure 18.7.1.5 of CESSAR-DC is labeled "PZR PRESS". Other apparent inconsistencies were noted with the use of the abbreviations SI and SIS within the IPSO display and the rest of DPS display hierarchy. These apparent inconsistencies conflict with guidelines from NUREG/CR-5908: 1.3.22 Abbreviation Rule and 1.3.3-4 Consistent Wording of Labels, which state that consistent abbreviations/labels should be used. This also conflicts with Section 2.2.2 of ABB-CE's HFESGB document, which states that abbreviations and acronyms should be unique. ABB-CE was requested to provide a justification for the current implementation of abbreviations or provide a commitment to make the necessary modifications.

Evaluation: ABB-CE stated that there is an abbreviation and acronym algorithm in the HFESGB. TOI entry No. 88 is tracking the issue of establishing consistent abbreviations and label conventions for System 80+. Hence, this issue is resolved.

(14) Issue: The DPS menu pages such as for CFM and PRI do not have titles associated with them. This is in conflict with Guideline 1.1-4 of the NUREG/-CR 5908. The absence of unique titles may result in confusion between menu pages and the top-level display pages of the corresponding plant sector. For example, the menu page for the primary coolant side of the plant, which is accessed by pressing “PRI” on the main menu bar, may be confused with the top-level display page, which is labeled PRIMARY (PRI) Level 1. ABB-CE is requested to provide a justification for the current implementation or provide a commitment to make the necessary modifications.

Evaluation: ABB-CE has entered Item 89 into its HF issue tracking system to ensure that titles are considered for all display pages, including the CFM and PRI menu pages. Hence, this issue is resolved.

(15) Issue: Inconsistencies were noted between the IPSO and the DPS CFM menu display with respect to the use of abbreviations and the location of critical function designators. The abbreviations RxC and SF on the IPSO apparently correspond to RC and SF on the DPS CFM menu display. This is inconsistent with guidelines from draft NUREG/-CR-5908: 1.3.2-2 Abbreviation Rule and 1.3.3-4 Consistent Wording of Labels, which state that consistent abbreviations/labels should be used. This

also conflicts with Section 2.2.2 of ABB-CE's HFESGB document, which indicates that abbreviations and acronyms should be unique. The locations of the individual critical function designators within the IPSO critical function matrix do not correspond well to their locations within the DPS CFM menu display. This is inconsistent with guidelines pertaining to consistency in the position of displayed information.

Evaluation: ABB-CE stated that TOI entry No. 88 has been made to establish consistent abbreviation/label conventions for System 80+. Once established, abbreviation consistency will be a straightforward implementation issue. TOI entry No. 90 has been made to confirm the consistency of and evaluate possible changes in the CFM scheme. The locations of the individual CFM designators on the IPSO critical function matrix (a 3 X 3 matrix) is not consistent with the CFM menu page layout because the CFM menu layout is consistent with the other menu formats (available CRT pages listed under a title). The IPSO and CFM 3 X 3 matrix saves space and corresponds to the CFM matrix convention. However, revision to the IPSO matrix will be considered.

This response was found acceptable based on ABB-CE's commitment to re-examine consistency between the IPSO and DPS when the HSI design is more complete. Consistency between the IPSO and DPS will also be addressed by the suitability verification with the final design.

18.6.2.3.2.6 Application of Design Method to Other System 80+ Control Room Panels

Section 18.7.4 of CESSAR-DC discusses the panels in the MCR other than the RCS panel. A number of positive features were noted such as functional grouping on panels, consistent use of System 80+ standard techniques and conventions, continuous display of RG 1.97 variables, and a separate communications panel. The following issues were identified.

(1) Issue: There does not appear to be a system level actuation in Section 18.7.4.2 of CESSAR-DC for the auxiliary feedwater system per IEEE-279.

Evaluation: ABB-CE stated that System 80+ emergency feedwater system level actuation is identified in CESSAR--DC Table 7.3-2, and shown in Figures 7.3-1c and 7.2. Conformance to IEEE-279 is described in Section 7.3.2.3-.2. System level activation is at the Plant Monitoring and Safety Monitoring Panels (two channels on each panel); detailed panel designs are to be determined for these panels. Hence, this issue is resolved.

(2) Issue: There may not be adequate communications coverage between the CR and auxiliary operators throughout the plant. There are no criteria or commitments in Section 18.7.4.13 of CESSAR-DC for full coverage, and there is no radio communications system as has been established by many current plants to address this issue.

Evaluation: ABB-CE stated that CESSAR-DC Section 18.7.4.13 (the communications panel) has been removed from Amendment N. The design of the plant Communications System as discussed in CESSAR-DC Section 9.5.2 (Amendment L) is current. Communications criteria are covered in HFESGB Section 6. Radio systems can be flexibly configured at COL applicant discretion; their incorporation in the design is not itself a requirement.

Section 9.5.2 of CESSAR-DC discusses the site telephone system, the public address system, sound-powered phone systems, offsite communications, and an intraplant portable, wireless communication system. CESSAR-DC commits to ensuring clear intelligible communications throughout the plant. Emergency power is provided for the wireless system. HFE aspects of the communications systems are appropriately addressed in Section 6 of the HFESGB document. This acceptably addresses this issue.

(3) Issue: The message tile monitor shown on Figure 18.7.4-3 for the feedwater and condensate system is not described. Is this actually a DIAS alarm tile display?

Evaluation: ABB-CE stated that the message tile monitor shown on Figure 18.7.4.3 was actually a DIAS alarm tile display that was incorrectly labeled. Hence, this issue is resolved.

(4) Issue: Section 18.7.4.5 of CESSAR-DC states that the CCS module of the safety monitoring panel provides access to all CCS controls and indications. (This capability is also stated for the remote shutdown panel.) The number of controls that may be accessed through the module may be large and impose high demands on the operator for control access and status monitoring. ABB-CE was requested to describe this module in greater detail along with provisions for facilitating control access and status monitoring.

Evaluation: ABB-CE stated that the design for the operator's module is not complete. ABB-CE entered Item 97 into its HF issue tracking system to ensure that the demands on the operator for control access and status monitoring are addressed during suitability analysis. Hence, this issue is resolved.

(5) Issue: Section 18.7.4.14 of CESSAR-DC describes the CRS console. The following issues were identified.

- The method by which work space requirements were identified is not described. ABB-CE was requested to describe the basis for the proposed design.

- Two potential benefits of including two DPS terminals in the CRS console are to compensate for the absence of dedicated (DIAS) indications and to allow rapid cross-checks to be made between different DPS display pages. These potential benefits are mitigated by the location of the DPS terminals at opposite ends of the console. ABB-CE was requested to provide its rationale for the CRS console.

Evaluation: ABB-CE stated that the CRS console is basically a desk that includes certain data processing and communications devices. It is specifically not a control panel, and does not support a specific set of rule-based tasks. Rather, supervisors perform less well-defined tasks that involve observing, reading, writing, data entry, and communications. The CRS console provides appropriate devices and generous space for two individuals (e.g., CRS and Shift Technical Advisor) to engage in such activities. The dual DPS screens present a tradeoff. If centrally located, their height impedes observation and communication; if located together at either end, their use by separate individuals would be restrictive. While facilitating side-by-side comparisons would be useful, it was deemed the less useful alternative. Continued evaluation of this issue will be considered, and has been entered as TOI Item 98. Hence, this issue is resolved.

(6) Issue: It was recommended that both the NRC I&C Branch and the Systems Branches review Section 18.7.4 since it contains material pertinent to their review areas.

Evaluation: The staff stated that ABB-CE should complete a consistency check between CESSAR-DC Section 18.7.4 and CESSAR-DC Chapter 7. ABB-CE agreed (Ref. 13 of CESSAR-DC Section 18.10, LD-93-147) to review the CESSAR-DC for consistency. Therefore, this issue is resolved.

18.6.2.3.2.7 Application of the Design Method to the Remote Shutdown Panel and Local Control Stations

(1) Criterion: HFE PRM General Criterion 3 states that the HSI design shall utilize the results of the task analysis and the I&C inventory to assure the adequacy of the HSI.

Evaluation: LD-92-120 states that local control stations required to perform emergency operations guidelines are designed using task analysis. Section 18.8.1 of CESSAR-DC states that the same human engineering criteria as that used for the MCR will be used for the remote shutdown panel. The HFPP and its subordinate documents make the ABB-CE commitments to the method explicit. Hence, this issue is resolved.

(2) Criterion: HFE PRM General Criterion 4 states that the HSI and working environment shall be adequate for the human performance requirements it supports. The HSI shall be capable of supporting critical operations under the worst credible environmental conditions.

Evaluation: ABB-CE was requested to describe provisions in the design process that will ensure adequate human performance in areas outside of the MCR where extreme environmental conditions such as high noise and requirements for protective clothing may exist. ABB-CE stated that the System 80+ maintains normal conditions for occupied workspaces as its design basis. Treatment of specific environmental hazards, particularly for local control stations, is provided via task analysis for tasks that are addressed by the emergency procedures. Task analysis is an acceptable method for assessing environmental factors that may affect task performance. Therefore this criterion is satisfied.

(3) Criterion: HFE PRM General Criterion 6 states that the selection and design of HSI hardware and software approaches shall be based upon demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options.

Evaluation: Sections 18.7.2.4 and 18.8.1.2 of CESSAR-DC state that the remote shutdown panel design uses the same panel profile as the main control console. It also uses the same criteria for human engineering and for information display and control allocation as the MCR. A review of criteria for selection and design of HSI hardware and software approaches are addressed in the review of MCR. Hence, this issue is resolved.

(4) Criterion: HFE PRM General Criterion 7 states that the HFE standards shall be employed in HSI selection and design. Staffing assumptions for the remote shutdown panel and the local control stations are important considerations for the application of these criteria. ABB-CE was requested to describe its staffing assumptions for the remote shutdown panel and the local control stations.

Evaluation: ABB-CE stated that staffing assumptions for the remote shutdown panel and the local control stations will be provided on a task-specific basis via FTA, consistent with the 10 CFR 50.54 and RG 1.114 staffing requirements. Hence, this issue is resolved.

(5) Criterion: HFE PRM General Criterion 9 states that HFE shall be applied to the design of equipment and software for maintainability, testing, and inspection. In particular, the following are not clearly described: (1) provisions for maintenance at locations in the plant such as local control stations, and (2) provisions for in-service, surveillance testing in the remote shutdown panel, local control stations, and other locations in the plant.

Evaluation: Maintainability considerations are provided in the HFESGB. ABB-CE stated that the design of HSI for surveillance testing are not complete, and therefore, cannot be described in detail. However, they will conform to the HFESGB. Selected surveillance and local control stations tasks will be addressed by FTA and validation. Hence, this issue is resolved.

Design Issues Related to Remote Shutdown Panel and Local Control Stations

(1) Issue: Section 18.8.1.4 of CESSAR-DC states the assumption that a reactor trip is performed prior to evacuating the MCR. Does the remote shutdown panel (RSP) provide the capability to perform a reactor trip? If not, where outside of the CR is the reactor trip capability provided?

Evaluation: ABB-CE stated that the remote shutdown panel does provide the capability to perform a reactor trip. Hence, this issue is resolved.

(2) Issue: Table 18.8-2 of CESSAR-DC indicates five alarms related to the primary coolant system that will not be provided on the RSP alarm panels because these alarms pertain to conditions that are ". . .not considered to occur coincidentally with control room evacuations." Since CR habitability is largely a separate concern from other plant failures and may occur for a variety of reasons, the rationale for excluding alarms for this subset of plant failures is unclear. ABB-CE was requested to describe its rationale in greater detail, including a discussion of how operators would cope with these failures in the absence of these alarms.

Evaluation: ABB-CE stated that credit is taken for considering MCR evacuation to be an uncomplicated scenario. The limited panel real estate available for dedicated displays will be allocated first to the credited safe shutdown success path applications. However, the fact that dedicated tiles are not provided does not mean that the alarms are not available. All alarms are available on the DPS at the RSP.

This concern is satisfied because alarms are available on the DPS at the RSP.

(3) Issue: Table 18.8-2 of CESSAR-DC states that the RSP will use a single RCP trouble alarm tile instead of the 16 RCP dedicated alarm tiles and the two seal/bleed alarm tiles that are provided on the RCS panel in the MCR. This appears to conflict with good design practice for alarms as reflected in EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development," which states, "Use of shared, or so-called "trouble" annunciator tiles should be minimized" and NUREG-0700 which states, "Annunciators with inputs from more than one plant parameter set point should be avoided." ABB-CE was requested to describe its rationale in greater detail. What are the implications for operator workload for processing alarms? How will this affect the operator's ability to rapidly determine the state of the RCPs when multiple alarm states have been tripped? What provisions have been made to ensure that the many alarm states associated with the RCPs will not interfere with the operator's ability to access other alarm information from the DIAS alarm tile display, including when using the alarm list displays?

Evaluation: ABB-CE stated that the Nuplex 80+ tile reduction philosophy acknowledges the paradox between the benefits of spatial dedication and the hazards of information overload in conventional control room alarm displays. Nuplex 80+ uses prioritization, functional organization, and digital technology to make alarm handling more manageable. Dedicated tiles now provide organizing and directing functions, but alarm information is provided through more flexible and dynamic messaging features. Lowest priority alarms are segregated from the high priority dedicated tiles. Guidance document caveats regarding multiple alarm inputs to single tiles are not applicable to the Nuplex 80+ implementation. These guidelines are concerned with the effort required to resolve the ambiguity of the alarm's source on conventional tiles. This is not an issue for DIAS because it provides individual messages.

ABB-CE subsequently agreed to address alarm system concerns through additional testing using prototypes of the DIAS alarm tile display system. Item 101 was entered into ABB-CE's HF issue tracking system to provide a commitment to conduct this testing. Sub-Item g of Item 101 addresses concerns related to the use of multiple alarms and the ability of operators to access alarm information.

Additionally, ABB-CE removed table 18.8-2 from CESSAR-DC and modified section 18.8.1.4 to indicate that RSP alarm requirements will be identified as part of the functional task analysis and detailed panel design. Hence, this issue is resolved.

18.6.2.4 Methods and General Characterisitics Findings

This review addressed:

- The methods for implementing the display and control requirements, selecting hardware and software, and refining of design concepts
- Design criteria used to determine CR and control panel arrangements including the overall configuration of the main control console and the position of individual control/display devices within individual panels
- General design characteristics that were incorporated into the HSI

These considerations were evaluated within the context of the MCR configuration, the presentation of information on controls and displays, and the layout of panels. Specific attention was given to the RCS panel and the remote shutdown panel.

This review found the application of methods, design criteria, and general design characteristics to be acceptable. Specific concerns identified included information presentation, panel layout, and configuration. ABB-CE provided responses and commitments via its HF issue tracking system to address these concerns in later stages of the design process. The most significant of ABB-CE's commitments was to provide more detailed descriptions of the human-system interface to support the following

- data entry tasks
- blocking and tagging tasks via the DPS and the DIAS of instrumented and non-instrumented components
- operator established alarms
- CCS operator module

In addition, ABB-CE committed to

- Establish consistent abbreviation conventions to be used throughout the System 80+ design.
- Evaluate the need for titles for all DPS display pages, including menu pages.
- Evaluate consistency between the CFM matrix configuration and the CFM menu page layout.
- Provide, in Amendment V of CESSAR-DC, illustrated examples of information coding of operator aid information for the DPS (TOI issue 100). This is part of FSER Confirmatory Item 1.1-1.

With respect to panel layout, ABB-CE also provided commitments via its HF issue tracking system to

- Evaluate the adequacy of laydown space in the controlling workspace for procedures and other materials.
- Evaluate the placement of DPS CRTs of the CRS console to determine whether side-by-side positioning or the inclusion of additional CRTs is necessary to support CRS tasks.

In addition, specific plant parameters cited by ABB-CE for inclusion in the System 80+ HSI were reviewed and approved by the staff.

18.6.3 Human Factors Engineering Standards, Guidelines, and Bases

The following is a review of documentation prepared by ABB-CE that provides HFE design criteria. Also addressed are (1) DSER issues that are relevant to HF guidelines, and (2) issues that were deferred from other human factors reviews associated with HFE PRM Element 6.

18.6.3.1 Objectives

The objective of this review is to evaluate ABB-CE's HFE principles and criteria used to identify, select, and design equipment to be operated/maintained/controlled by plant personnel with respect to accepted HF guidance and practices. This review focused on the specific HF principles and criteria that ABB-CE identified as a basis for the System 80+ HSI and embodied in the ABB-CE document, "Human Factors Engineering Standards, Guidelines, and Bases" (HFESGB). The HFESGB document was evaluated with respect to its scope, technical basis/validity, level of detail, and procedure for implementation.

18.6.3.2 Methodology

18.6.3.2.1 Description of Review Methodology

Reviewed the HFESGB document as a whole for scope, technical basis and validity, level of detail, and procedure for implementation. Evaluated the overall technical basis of the document by reviewing the source documents referenced in the HFESGB. Evaluated individual guidelines on a selective basis for technical basis/validity and level of detail. Reviewed with ABB-CE discrepancies or concerns identified during this review and subsequently resolved through clarification, modification, or inclusion in ABB-CE's HF TOI with a commitment to address the issue more fully at a later stage in the design process.

18.6.3.2.2 Material Reviewed

The following ABB-CE documents were referenced in this review:

- Reference 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 1, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992.

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18, DSER Open Item Response," ABB-CE letter dated January 18, 1993.

- Reference 4 of CESSAR-DC Section 18.10, LD-93-135, "System 80+ Information for Issue Closure," Attachment 1, "ABB-CE Response to System 80 Operating Experience Issues Based Upon Interviews with System 80 Operators," ABB-CE letter dated September 1, 1993.

- Reference 5 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 5, "SSAR-DC Markups for V&V and Procedures," ABB-CE letter dated September 24, 1993.

- Reference 6 of CESSAR-DC Section 18.4, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+," (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).

- Reference 3 of CESSAR-DC Section 18.4, "Human Factors Engineering Verification and Validation Plan for Nuplex 80+," (NPX80-IC-VP790-03, Rev. 00, September 24, 1993).

- Reference 4 of CESSAR-DC Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 02, September 29, 1993).

- CESSAR-DC.

18.6.3.2.3 Design Criteria Documents

Consulted the following materials as part of this evaluation:

- American National Standards Institute, ANSI HFS-100, "American National Standard for Human Factors Engineering of Visual Display Terminal Workstations," 1988.

- DOD-HDBK-761A: Human Engineering Guidelines for Management Information Systems, 1990, (Department of Defense).

- EPRI NP-3659: Human Factors Guide for Nuclear Power Plant Control Room Development, 1984, (Electric Power Research Institute - Kinkade, R.G., and Anderson, J.).

- EPRI NP-3701: Computer-Generated Display System Guidelines (Vols. 1 and 2), 1984, (Electric Power Research Institute - Frey, R. et al.).

- EPRI NP-4350: Human Engineering Design Guidelines for Maintainability, 1985, (Electric Power Research Institute - Pack, R. et al.).

- ESD-TR-86-278: Guidelines for Designing User Interface Software, 1986, (Department of Defense).

- Gilmore, Walter E. et.al, User-Computer Interface in Process Control: A Human Factors Engineering Handbook, Academic Press, San Diego, CA, 1989.

- Human-Computer Interface Style Guide (Version 1), 1992, (Department of Defense - Defense Information Systems Agency).

- International Electrotechnical Commission (1989), "International Standard: Design for Control Rooms of

Nuclear Power Plants" (IEC-964), Geneva, Switzerland: Bureau Central de la Commission Electrotechnique Internationale.

- MIL-HDBK-759A: Human Factors Engineering Design for Army Material, 1981, (Department of Defense).
- MIL-STD-1472D: Human Engineering Design Criteria for Military Systems, Equipment and Facilities, 1989, (Department of Defense).
- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), attached response to DSER Item 20.2-29," ABB-CE letter dated December 18, 1992.
- U.S. Nuclear Regulatory Commission (1980), "Functional Criteria for Emergency Response Facilities" (NUREG-0696), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1981), "Guidelines for Control Room Design Reviews," (NUREG-0700), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1984), "Standard Review Plan (Rev. 1)," (NUREG-0800), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1992), Advanced Human-System Interface Design Review Guideline," (Draft NUREG/CR-5908), Washington, D.C.
- USE-1000: Space Station Freedom Human-Computer Interface Guide. Houston, TX: NASA (1988).

18.6.3.2.4 Scope and Limitations

The focus of this review is on HF guidelines identified by ABB-CE that pertain to their design of the HSI both within and outside of the MCR. The human factors guidance documented in HFESGB will be expanded and modified as HF issues are identified through the continued design efforts and as a result of the HF review process. Thus, the review is limited to guidance currently included in the document. Individual guidance was sampled conducted using a sampling methodology. This methodology to include a review of the coverage of guidance for the individual topics addressed by the HFESGB as well as detailed review of guidance for design characteristics that could be demonstrated by the MCC mockup. This provided a context for examining the application of the guidance.

18.6.3.3 Results

Results of this review are organized in three major sections: DSER review, general criteria review, and specific issues. The DSER review addresses those DSER issues that are most relevant to the guidance provided in the HFESGB document. The general criteria review addresses the following issues: scope, technical basis/validity, level of detail, and procedure for implementation. The specific issue review addresses specific concerns that were identified during resolution of the DSER issues, identified through sampling of individual guidelines, or deferred from other portions of the HF review process.

18.6.3.3.1 DSER Review

This section provides a review of those DSER issues that were most relevant to the HF guidance that was addressed by the HFESGB review. The following DSER issues were reviewed:

- 18.8-1 - Human-System Interface Design
- 18.8.1.5 - Quality and Types of Information Encoded in the Control Room
- 18.8.2 - Additional HSI Information Required for Staff Review

DSER Issue 18.8.2 contained 19 sub-items (see Section 18.6). The following sub-items were relevant to the HFESGB review:

(a) Provide human engineering justification for the number of colors, shapes, and patterns used to convey information in the CR (Sub-Item 7).

(b) Provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for:

(c) Readability of alarm text and tiles from all operator positions in the CR

(d) Number of colors and shades used on displays

(e) Types and amount of information encoded in the CR as well as the encoding techniques used

(f) Audible and tactile feedback for controls, controllers and other devices

These DSER items are addressed below.

18.6.3.3.1.1 DSER Issue 18.8-1 - Human-System Interface Design

Criterion: DSER Issue 18.8-1, Human-System Interface Design, states that ABB-CE does not address in CESSAR-DC Chapter 18 how the results of the human engineering systems analyses were applied to the selection or design of the System 80+ CR, control panels, software, and hardware.

Evaluation: This issue was held open pending resolution of all other DSER and HFE PRM issues related to the HSI design found in Sections 18.6.1 through 18.6.3. Based on the resolution of all other HSI design issues this issue was found to be acceptably addressed.

18.6.3.3.1.2 DSER Issue 18.8.1.5 - Quantity and Types of Information Encoded in the Control Room

Criterion: DSER Issue 18.8.1.5, Quantity and Types of Information Encoded in the Control Room, states, "ABB-CE should provide empirical data or other means of demonstrating that operators can effectively utilize the 27 properties and parameters encoded, the 15 colors, and the 10 shape/symbol codes under normal, abnormal, and emergency operations and under all modes of plant operations."

Evaluation: ABB-CE's discussion of coding schemes (Ref. 7 of CESSAR-DC Section 18.10, LD-93-005) justified the codes based on the number and types of code discriminations that operators must perform for specific information processing activities. In addition, the results of the suitability analysis, reported in Part C of NPX80-TE790-01, described a critique of specific coding characteristics that were found to be problematic during this analysis. While ABB-CE's responses provided a rationale for important characteristics of the coding scheme and provided evidence that the scheme has undergone refinement, they do not demonstrate the ability of operators to use the coding scheme under the various operating conditions that were specified in the DSER issue. The effectiveness of the coding scheme will be addressed in an operational setting during the integrated system validation portion of ABB-CE's V&V program. The various operating conditions specified in the DSER issue will be addressed at that time. ABB-CE's commitment in the HFPP and the V&V plan to perform this testing was found to address this issue satisfactorily. Hence, this issue is resolved.

18.6.3.3.1.3 DSER Issue 18.8.2.a - Human Engineering Justification

DSER Issue 18.8.2, Additional HSI Information Required for Staff Review, states, in part, that ABB-CE should provide human engineering justification for seven characteristics of the HSI design. The following is a review of Sub-Item 7 - the number of colors, shapes, and patterns used to convey information in the CR.

Evaluation: ABB-CE provided a justification (Reference 7 of CESSAR-DC Section 18.10, LD-93-005) for the coding scheme used to convey information in the CR. ABB-CE stated that alarm and information processing is context dependent; the number of codes that require discrimination is reduced to a manageable set for specific uses. The table "Nuplex 80+ Coding Matrix" describes information properties that are conveyed through shape, color, color intensity, flash rate, and switch position. Review of this table and supporting text indicated that the number of codes requiring discrimination is reduced by the context of the operator task. For example, identification of alarm priority requires discrimination of four shape codes: a solid (reverse video) box, an open box (frame), brackets, and underline. This is consistent with Guideline 1.3.9-7, Clearly Discriminable Shapes, of draft NUREG/CR-5908, which indicates that as many as 15 different shapes can be readily distinguished if the shapes are properly designed. The table indicates that eight color codes are used. However, the number of discriminations is limited to two or three depending on the context. For example, identification of symbols for active and inactive components require discrimination of the colors red and green while identification of labels for dynamic data and RG 1.97 Category 1 data requires discrimination of the colors cyan and purple. This is consistent with draft NUREG/CR-5908 Guideline 1.3.8-9, Minimum Color Differences, which states that at least 7 to 10 simultaneous colors may be discriminated if they are significantly different.

While the coding scheme was generally consistent with HF guidelines, the following issues were identified.

(a) In the table of Reference 7 of CESSAR-DC Section 18.10, LD-93-005, ABB-CE identifies three alarm states (e.g., unacknowledged, acknowledged, reset) corresponding to three intensities of yellow (e.g., bright, saturated/dull, and dark). These three alarm states conflict with those provided in design description documentation - unacknowledged, acknowledged, cleared. ABB-CE was requested to clarify this discrepancy.

ABB-CE's response in Reference 14 of CESSAR-DC Section 18.10, LD-93-135 provides revised terminology for both the alarm states and the intensity levels. The four alarm states are identified as: new, existing, cleared, and reset. Reset is the null state (i.e., no alarm) and is null coded - labeled but otherwise is an empty tile outline. The three relative intensity values are identified as high, medium, and low. ABB-CE stated that the revised terminology for alarm states and intensity levels will appear in a future amendment of CESSAR-DC. Table 18.7.1-1 of CESSAR-DC is an update of the table from Reference 7 of CESSAR-DC Section 18.10, LD-93-005. The terminology in this table was modified in Amendment V to CESSAR-DC. Therefore, this issue is resolved.

(b) The identification of alarm state requires the discrimination of three intensities of yellow. This is in conflict with Guideline 1.3.10-5 of draft NUREG/CR-5908 which states that coding by differences in brightness should be used for applications that only require discrimination between two categories of display items.

Based on a review of HF literature it was determined that the use of three intensity (brightness) levels does challenge the limits of acceptability provided by available HF guidance and, therefore, underscores the importance of testing. In addition, ABB-CE should take efforts to maximize the differences between the brightness levels. MIL-HDBK-761A states that each level of brightness coding should be separated from the next nearest level by at least a 2:1 ratio. Brightness levels selected for the alarm codes should be verified against this criteria. In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE agreed to enter this concern as Item 101 in its tracking system for open HF issues. Item 101 states that a number of concerns, including brightness coding, will be evaluated further using a prototype prior to V&V testing. ABB-CE also agreed to verify that the brightness levels in the final design vary by at least a 2:1 ratio. Hence, this issue is resolved.

(c) The coding for both active and inactive equipment status is indicated by the same switch position code (bottom) on both the CCS process controllers and the CCS switches. ABB-CE is requested to describe how the bottom switch position code is represented on the CCS process controllers and CCS switches and how the switch position code is used to identify equipment status.

In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE described switch positions and codes for CCS process controllers and CCS switches. In addition, Table 18.7.1-1 of CESSAR-DC was modified to indicate that the coding scheme for two-state components is as follows: active equipment is top and inactive is bottom. This concern was satisfied by the description of switch positions and codes and modification to CESSAR-DC.

(d) A color is not specified for the cross-hatch marks, which are used to indicate that a component is uncontrollable from the CCS. Should this color be considered a color code?

In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE stated that cross-hatching is applied as a texture without color and is therefore not a color code. ABB-CE's position that cross-hatch marks should not be considered a color code was found acceptable because the cross-hatch marks are applied differently than color and conveyed different types of information than the color codes. Therefore, this concern is satisfied.

18.6.3.3.1.4 DSER Issue 18.8.2.b - System 80+ Specific Studies

DSER Issue 18.8.2, Additional HSI Information Required for Staff Review, states, in part, that ABB-CE should provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for:

(a) Readability of alarm text and tiles from all operator positions in CR

(b) Number of colors and shades used on displays

(c) Types and amount of information encoded in the CR as well as the encoding techniques used

(d) Audible and tactile feedback for controls, controllers and other devices

The number before each item corresponds to the numbers used in Section 18.6. The Items 3, 5, 6, and 7 are discussed below.

(3) Readability of alarm text and tiles from all operator positions in CR

Evaluation: ABB-CE's response cited the following sections of HFESGB as the criteria for verifying readability, Section 2.5 - Equipment Labels and Section 5.3.2 Tile Matrices. The HFESGB criteria are based largely on accepted guidelines. The response did not present System 80+ specific studies. However, issues have already been entered into ABB-CE's tracking system for open HF issues to ensure that the effectiveness of the alarm tile coding scheme will be evaluated. Additional tracking items are not necessary. Hence, this issue is resolved.

(5) Number of colors and shades used on displays

(6) Types and amount of information encoded in the CR as well as the encoding techniques used

Evaluation: ABB-CE's response to these items indicated that the bases for these items were derived largely from existing guidelines rather that System 80+ specific studies. The suitability analyses, cited in ABB-CE's responses, were System 80+ specific studies that were performed to identify problematic features. However, these studies did not define thresholds of adequate and inadequate human performance. The effectiveness of these items will be addressed through validation studies. However, the validation studies will not define thresholds of acceptable human performance with respect to these specific issues.

The verification analysis will evaluate the acceptability of the coding scheme against HF guidelines and the validation studies will address acceptability with respect to the HSI as a whole. This approach was considered acceptable for evaluating the HSI because it will determine whether the coding scheme is consistent with human factors guidance and can be used effectively in the task environment. Thus acceptability can be established by determining that human performance is adequate without defining the thresholds of adequate and inadequate human performance.

(7) Audible and tactile feedback for controls, controllers and other devices

Evaluation: ABB-CE stated that auditory feedback is not used in System 80+ and tactile feedback is only provided for pushbuttons. ABB-CE stated that the criteria for verifying suitability of tactile feedback for pushbuttons were provided in Section 3.2 of HFESGB and Part C (Suitability Analysis) of NPX-TE790-01. Because the issue of tactile feedback for pushbuttons is largely understood and because the pushbutton hardware is commonly used in industry, the requirement for System 80+ specific studies is not considered necessary. However, ABB-CE was requested to provide its justification for not providing auditory feedback for touch screens.

In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE stated that auditory feedback would be redundant with the existing visual feedback and was considered unnecessary to ensure the effective usability of the interface. Although ABB-CE has not excluded consideration of its later use, ABB-CE does not consider the lack of redundant auditory feedback to be a deficiency in the design. The reviewers found ABB-CE's position to be acceptable because visual feedback is more immediate and salient than auditory or tactile feedback given the state of the art of touch screen interfaces. Direct tactile feedback is only possible through devices such as glove interfaces, which are not highly practical for NPP control applications. Auditory feedback provided via audio-speakers is less direct and less salient than visual feedback that is emitted from the touch target. In addition, auditory feedback has potential disadvantages that must be carefully considered including accidental activation of adjacent touch targets due to reliance on auditory feedback alone, and distraction/confusion resulting from auditory feedback from other touch screen devices. Hence, this issue is resolved.

18.6.3.3.2 General Criteria Review

18.6.3.3.2.1 Scope

A review was conducted to determine whether aspects of the HSI that are important to the safe operation and maintenance of the plant by personnel are addressed in the HFESGB document. The guidance included in HFESGB was compared against the topic areas presently addressed in draft NUREG/CR-5908, NUREG-0700, and other topics identified as important to safety. The draft HFESGB generally covered the topic areas included in NUREG-0700, which addresses HF issues related to traditional CRs. In addition, HFESGB addressed many, although not all, of the advanced HSI topics that are reflected in draft NUREG/CR-5908 and other guidelines. The following specific concerns were identified:

(1) Issue: Standard design features - HFESGB lacks guidance pertaining to the standard design features. The DIAS dedicated and multiple parameter displays, DIAS alarm tile displays, CCS process controller, and other standard design features, are to be used throughout the HSI.

Evaluation: While specific examples of the standard features are provided in CESSAR-DC for the RCS and CVCS panels, guidance is not provided in HFESGB for the general application of the standard features. ABB-CE agreed to provide in a future revision of the HFESGB an additional section that provides illustrations of each standard design feature that are annotated with references to relevant HFESGB guidelines for important characteristics. ABB-CE has entered Item 105 (Sub-Item a) into the TOI to record its commitment to provide this guidance at which time it will be reviewed by the staff. This is acceptable because the a final design is not available for design certification review. The relevant guidance will address detailed design features which will be developed later in the design process as the design matures.

(2) Issue: Graphic formats - HFESGB lacks specific guidance pertaining to graphic formats used in the DPS displays including mimics, bar charts, deviation bar charts, etc. Relevant guidance including graphic orientation, labeling, coding, etc. should be addressed.

Evaluation: ABB-CE agreed to provide in a future revision of the HFESGB specific additional guidance pertaining to graphic formats used in DPS displays such as bar graphs, mismatch bar graphs, deviation bar graphs, schematic displays, and tables. This guidance will address the following where appropriate: the selection of display formats and descriptions of graphic orientation, descriptor and title conventions, and coding. ABB-CE has entered Item 105 (Sub-Item b) into the TOI to record its commitment to provide this guidance at which time it will be reviewed by the staff. This is acceptable because the a final design is not available for design certification review. The relevant guidance will address detailed design features which will be developed later in the design process as the design matures.

(3) Issue: Data entry fields and human-computer interaction - HFESGB lacks specific guidance pertaining to the design of the human-computer interface associated with entering and maintaining data associated with operator established alarms, component, tagging and blocking, and status indication of non-instrumented components.

Evaluation: ABB-CE agreed to provide in a future revision of the HFESGB additional guidance pertaining to data entry/text editing tasks appropriate to operator established alarms, component tagging and blocking, and entry of non-instrumented component status data. ABB-CE entered Item 105 (Sub-Item c) into the TOI to record its commitment to provide this guidance at which time it will be reviewed by the staff. This is acceptable because the a final design is not available for design certification review. The relevant guidance will address detailed design features which will be developed later in the design process as the design matures.

(4) Issue: Guidance for extreme environmental conditions - The HFESGB does not provide design guidance pertaining to the operation and maintenance of plant equipment in areas of extreme environmental conditions such as high noise or contamination/radiation that may require wearing protective equipment (e.g., ear protection, respirators, gloves, anti-contamination clothing).

Evaluation: ABB-CE stated in Reference 2 that the System 80+ maintains normal environmental conditions as its design basis for occupied workspaces. In lieu of providing general guidance for extreme environmental conditions, environmental hazards are treated on an ad hoc basis in response to the results of task analysis. Task analysis will be performed for operator tasks performed outside of the CR (e.g., local control stations) that are addressed by emergency procedures. ABB-CE's position that guidance for extreme environmental conditions were not included in the HFESGB document because extreme conditions are not consistent with the design basis was found acceptable. The task analysis and the method for addressing environmental conditions was reviewed in Section 18.5 and found to be acceptable. Therefore, this concern is resolved.

(5) Issue: Surveillance testing - Provisions for surveillance testing of the MCR panels, the remote shutdown panel, local control stations, and other locations in the plant have not been clearly described.

Evaluation: While the HFESGB does not include a section dedicated to surveillance testing, the maintenance section of the HFESGB incorporates good HF principles which would support surveillance activities. Hence, this issue is resolved.

18.6.3.3.2.2 Technical Basis/Validity

The content of design-specific guidelines and specifications should be derived from (1) the application of generic HFE guidance to the specific application, and (2) the development of the designer/applicant's own guidelines based upon design-related analyses and experience.

Selection of generic HFE guidelines documents as sources for design-specific guidance should be based upon consideration of "validated" principles. Validity may be defined in terms of two aspects

- the degree to which the individual guidelines within a source document are based upon empirical research and an audit trail from each guideline back to its basis
- the degree to which the source document is subjected to peer review

In general, documents which satisfy both of these criteria are considered the best primary source documents for a design-specific application.

Design guidelines/specifications may contain guidance that is not derived from generic HFE guidelines and may contain guidelines that are discrepant from NRC review guidance. In these cases, justification should be provided to support the review of the acceptability of these guidelines. A documented analysis-based rationale should be provided for these guidelines, such as

- an analysis of recent literature
- an analysis of current practices
- tradeoff studies and analyses
- the results of design engineering evaluations

The following are the results of a review of the HFESGB document based on these considerations.

(1) Issue: Validity of generic guidelines.

Evaluation: The reference section of the HFESGB stated that the guidelines were derived from 26 source documents. This list states many of the significant works in HSI guidance. This list includes seven of the eight source documents of draft NUREG/CR-5908 that have undergone extensive review. It also included an additional document that was included in the list of recommended source documents in HFE PRM Element 6 General Criterion 11. These eight source documents are frequently cited in Part B, the bases section of HFESGB. The HFESGB list includes additional documents such as handbooks (Helander, 1988 and Salvendy, 1982), textbooks (Tufte, 1983), and journal articles (Ledgard, 1989). These sources were not included in draft NUREG/CR-5908 or HFE PRM Element 6 General Criterion 11 because their validity was considered more difficult to establish. However, use of these additional documents was acceptable because rationales were provided for the specific guidelines derived from these sources. Hence, this issue is resolved.

18.6.3.3.2.3 Level of Detail

Generic HFE guidelines should not be used in the abstract. The tailoring (translating/interpreting) of individual guidelines to the specific design through function and task analysis data should be reflected in the designer/applicant's document and should be available for review. The designer/applicant's document should be detailed enough to permit use of the document by design personnel and subcontractors to achieve a clear, consistent, and verifiable design that meets the designer/applicant's guideline/specification.

The following issue is the result of a review of the HFESGB document based on these considerations.

(1) Issue: System 80+ versus generic guidance - The HFESGB should provide specific guidance that have been extracted from the broad body of existing HF literature and other sources to provide rationales/justifications for specific aspects of the System 80+ design. The HFESGB should also include general guidance to support design decisions that have not yet been made.

Evaluation: A review of the HFESGB indicated that general guidance is provided when specific guidance would seem more appropriate. Requirements for specific guidance were addressed in Issues 1, 2, and 3 of Section 4.2.1. ABB-CE has provided a commitment through Item 105 of the TOI to provide this additional guidance at which time it will be reviewed by the staff. This is acceptable because the a final design is not available for design certification review. The relevant guidance will address detailed design features which will be developed later in the design process as the design matures.

8.6.3.3.2.4 Procedure for Implementation

The designer/applicant's guideline specification document should provide an indication of how it is to be used in the overall design process.

The following are the results of a review of the HFESGB document based on these considerations.

(1) Issue: Procedures that ensure systematic application of guidance to display design - While the HFESGB provides guidance regarding the details of display design, it does not provide guidance to designers to ensure systematic application of its guidelines.

Evaluation: ABB-CE provided the following commitment to apply a systematic process to display design in Appendix A, Section A-3.5.2.1.6 of the HFPP:

> The reference design for the MCR and Remote Shutdown Room (RSR) indications and controls (i.e., screen design, panel layout, etc.) shall be detailed through a systematic process incorporating HFE design guidance. Appropriate documentation for the systematic process shall include the following (1) documentation showing the results of design reviews, (2) documentation that shows how the results of the functional task analysis are being applied to the design of specific displays, and (3) a checklist for each display page indicating important characteristics.

This response satisfactorily addresses the intent of this issue by specifying specific steps that will be performed and documented as part of the design process. Although detailed guidance for a systematic design process is not provided, this commitment will insure that the design process proceeds systematically and that documented evidence of a systematic approach will be available. Therefore, this concern is resolved.

(2) Issue: Procedures for HSI design in non-CR environments - Neither the HFESGB nor the FTA methodology provides a procedure or guidance for systematically reviewing environmental concerns.

Evaluation: ABB-CE commits to using task analysis as an input to the design of local control stations that are addressed by the emergency operations guidelines. Provisions have been made for recording significant environmental considerations in a miscellaneous category. The evaluation of environmental considerations will be addressed by ABB-CE via suitability verification during verification and validation. Hence, this concern is resolved.

(3) Issue: Procedures for establishing the precision with which values are displayed - A review of the design documents and the HSI mockup indicates the apparent lack of a systematic approach for determining the degree of precision with which data are presented, whether in digital or graphic form, to the operator via the HSI.

Evaluation: Section 18.5.1.5.3 of CESSAR-DC states that parametric requirements for display and control variables will be defined in terms of device type, range, accuracy, and units of measure as part of the FTA methodology. While accuracy of data is an important requirement, it is a separate concern from the precision with which data is presented to the operator (e.g., the number of significant digits in digital displays, the number of intervals on scale displays). Specific examples of the lack of clearly defined display precision requirements were observed during the design features review with respect to the scaling on bar charts and other indicators. The HFESGB provides general criteria for scaling. However, this is insufficient for determining the precision requirements for specific parameters.

In Amendment S to the CESSAR-DC, ABB-CE modified Section 18.5.1.5.3 to require that parametric requirements for display and control variables be defined in terms of precision, in addition to, device type, range, accuracy, and units of measure. This entry states that the display precision of each measured variable is provided based on operator task requirements. In addition, ABB-CE modified Section 6.1.5.2 - Phase 2 Availability Inspection Criteria of its V&V plan (NPX80-IC-VP790-03) to state that precision specifications will be verified for each as-built item of the HSI. These modifications satisfy the review issue.

18.6.3.3.3 Specific Issues

This section provides a review of issues and concerns that were identified through reviews of selected guidelines of the HFESGB or were identified during the evaluation of other review issues.

(1) Issue: Symbols and graphical formats - Graphic forms (e.g., bar charts and deviation bar charts) that are used in the HSI displays and controllers are not adequately described in the HFESGB.

Evaluation: The symbols and graphical formats included in HFESGB were found to be generally consistent with those used within the nuclear power industry. The specific implementation of these symbols and graphical formats will be evaluated during verification and validation when the design is complete. However, the review indicated that the specific graphic forms (e.g., bar charts and deviation bar charts) that are used in the HSI displays and controllers are not adequately described in the HFESGB.

ABB-CE agreed to provide in a future revision of the HFESGB specific additional guidance pertaining to graphic formats used in DPS displays such as bar graphs, mismatch bar graphs, deviation bar graphs, schematic displays, and tables. ABB-CE entered Item 105 (Sub-Item b) into the TOI to record its commitment to provide this guidance at which time it will be reviewed by the staff. This is acceptable because the a final design is not available for design certification review. The relevant guidance will address detailed design features which will be developed later in the design process as the design matures.

(2) Issue: The criteria for minimum character heights specified in HFESGB is inconsistent with draft NUREG/CR-5908 and other guidelines.

Evaluation: Guideline 2.2.3.2b of HFESGB states a recommended character height of 18 to 20 subtended MOA at the design basis reading/working distance and a minimum of 12 MOA at an unspecified distance to ensure legibility. A review of the basis for Guideline 2.2.3.2b found in Part B of HFESGB indicates that ABB-CE had selected 12 MOA as a "robust" standard for minimum character height. These values were in conflict with Guideline 1.3.1-4 of draft NUREG/CR-5908, which recommends a minimum character height of 16 MOA and a maximum of 24 MOA.

A subsequent review of ABB-CE's rationale for using 12 MOA as a "robust" minimum value was conducted. A review of HF literature indicated recommended character heights in access of 12 MOA. The conditions for which smaller character heights may be appropriate (e.g., where reading speed is not important, where sufficient contrast is provided) were considered not relevant to this CR application.

ABB-CE applied the criterion value of 12 MOA and predetermined maximum viewing distances to determine minimum character heights, in centimeters (in.), for text presented on control panels. Minimum character height values, in centimeters (in.), were determined for three categories: text viewed at the panel, text viewed from an adjacent panel, and text viewed from across the MCC. These values are shown in Table 2.2.3.2 of HFESGB.

It was recognized that actual viewing distance and display hardware are important factors in display legibility. The actual operator viewing distances may be different from (less than) the maximum viewing distances specified in Table 2.2.3.2 of HFESGB. The final display hardware may be different from that used in the prototype. ABB-CE was requested to consider specifically evaluating text legibility under conditions that are representative of actual use. ABB-CE agreed and entered issue 102 into the TOI to ensure that the legibility of controls and displays will be evaluated under conditions that are representative of anticipated work conditions. Hence, this concern is resolved.

(3) Issue: The criteria for minimum size of poke areas for touch screens specified in HFESGB is inconsistent with draft NUREG/CR-5908 and other guidelines.

Evaluation: Guideline 3.4.9.1 of HFESGB specifies that touch target areas should have a minimum height of 6 mm (0.25 in.), a minimum area of 161 mm^2 (0.25 sq in.) and a resulting minimum width of 25 mm (1.0 in.). In addition, the HFESGB criteria for separation of touch target areas is unclear. Guideline 3.2.4-10 of draft NUREG/CR-5908 specifies a minimum height and width of 15 mm (0.6 in.) with a resulting minimum area of 232 mm^2 (0.36 sq in.). Therefore, the size recommended by draft NUREG/CR-5908 is over 40 percent larger than the area specified by HFESGB. ABB-CE was requested to address this apparent inconsistency.

ABB-CE's basis for Guideline 3.4.9.1 of HFESGB was subsequently reviewed in greater depth including consideration of unique characteristics of the System 80+ touch screens such as the provision of visual feedback when the touch area is entered and the "make on break" mode of actuation. Based on these considerations the minimum touch area dimensions were found to be acceptable because the visual feedback of touch area and the "make on break" mode of actuation reduced the likelihood of accidental activation. In addition, ABB-CE agreed to clarify the wording of the criteria for separation of touch target areas. Therefore, this concern is satisfied.

(4) Issue: Anthropometric dimensions.

Evaluation: The bases for HFESGB Guideline 7.5.2.1 provides a discussion of anthropometric data pertaining to the distance from the central axis of the body to the panel edge and the eye distance forward of the central axis of the body. However, the use of these dimensions in the evaluation of panel dimensions is not clear in this discussion.

ABB-CE's response in Reference 2 explained how these dimensions may be used in the evaluation of panel dimensions. Therefore this concern is satisfied.

(5) Issue: Justification of vision and reach envelopes on System 80+ control panels.

Evaluation: The description of control panel dimensions found in Section 18.6.5.7 of CESSAR-DC states that the anthropometric data for these profiles are based on the HFESGB and MIL-STD-1472D. Further justification for the specific dimensions of these panels was not clear based on the material presented in Section 7.6.2.1 of HFESGB. This section states that the reach envelopes are unique to each panel according to bench board depth and slope and must be evaluated individually. Based on this review the following concerns were identified.

(a) Figures 18.6.5-11 and 18.6.5-12 of CESSAR-DC show eye heights for a 95th percentile male and a 5th percentile female. This position is aligned with the leading edge of the bench board. Why are these positions not set off horizontally to allow for torso and head width? What are the resulting maximum viewing distances for the male and female? How do they compare to the specified viewing distances for controls and displays?

ABB-CE's response in Reference 2, LD-93-138, "System 80+ Information for Issue Closure, Attachment 12, sub-attachment 1, ABB-CE Responses to Human Factors Engineering Standards, Guidelines, and Bases Technical Evaluation Report" states that the "at-the-panel" viewing distance in HFESGB Section 2.2.3.2 (i.e., 91 cm (36 in.)) is only slightly greater than the 95th percentile male reach envelope, which is shown to easily capture the panel work surfaces. Maximum viewing distances imposed by the panels, which are most limiting for the 5th percentile female, are less than 76 cm (30 in.) for standup panels, less than 79 cm (31 in.) for sitdown panels. This provides an adequate margin for torso width, particularly since nothing except oversize labels will be available for reading on the uppermost panel edges.

This explanation was found to be satisfactory. Hence, this concern is resolved.

(b) Figures 18.6.5-11 and 18.6.5-12 of CESSAR-DC show arm reach for a 95th percentile male and a 5th percentile female. These figures indicate that a 5th percentile female cannot reach the upper portion of the stand-up panel when standing nor the upper portion of the sitdown panel when sitting. ABB-CE was requested to provide justifications for these apparent design discrepancies.

ABB-CE's response in Reference 2 states that tradeoffs between panel area, cabinet access, frequency of use, and 5th percentile female stature dictate that seated operators may be required to stand to acknowledge some DIAS alarm tiles on the MCC; similarly, on ACSC panels, a standing operator may be required to use the DPS for alarm acknowledgement. All control devices are within the seated reach envelope. Panels could even be made taller, if viewing and visibility were not impacted. The designs are thus sufficient to support task demands.

This explanation was found to be satisfactory since the design does not prevent operators from performing necessary tasks. Therefore, this concern is resolved.

18.6.3.4 Findings

This review indicated that the HFESGB

- has an acceptable scope that includes aspects of the HSI, both inside and outside of the MCR, that are important to safe operation and maintenance of the plant by personnel
- included general design guidance that was derived from acceptable HF source documents

The guidance provided by the HFESGB was presented at a level of detail that was appropriate for many of the design areas addressed. However, in some cases specific guidance was lacking with respect to unique aspects of the System 80+ HSI design. Three areas were identified

- standard design features
- graphic formats used in DPS displays
- data entry/text editing

ABB-CE committed to include additional guidance in the HFESGB to address these three areas. Guidance for the standard design features will include illustrations annotated with references to relevant guidelines for important characteristics. Guidance for the graphic formats will include topics such as criteria for selection of formats, descriptions of graphic orientation, descriptor/title conventions, and coding. Guidance for data entry/text editing will address operator established alarms, component tagging and blocking, and entry of status information for non-instrumented components.

The review identified a lack of procedures or other guidance for the systematic implementation of the HFESGB guidelines and standards for the design of DPS displays. While procedures for other design activities such as CR layout and panel layout are well defined in CESSAR-DC, neither the HFESGB nor CESSAR-DC provided similar procedures for the application of HFESGB guidance to the design of DPS displays. ABB-CE addressed this concern by providing a commitment in its HFPP to employ a systematic process for display design. The application of this systematic process can be verified through documentation showing the results of design reviews, the application of FTA results, and checklists of important characteristics for each display page.

Through the sampling review of detailed guidelines, some significant differences were found between specific design criteria provided by the HFESGB and design criteria recommended by draft NUREG/CR-5908. An example is the design criteria for minimum character height. In this particular case, the discrepancy was addressed through a commitment by ABB-CE to specifically evaluate legibility in the design. Since (1) the guidance of the HFESGB must be interpreted within the context of the actual System 80+ design to ensure compatibility with the underlying assumptions of tasks and environment that the HFESGB source documents were based upon, and (2) the System 80+ design is the subject of an ongoing design process, the verification and validation process will address both the details of the HFESGB guidance and the interpretation of this guidance within the context of the detailed HSI design.

The commitments made by ABB-CE in response to this review adequately address the general concerns of scope, technical basis/validity, level of detail and procedure for implementation that were this review's focus. On that basis the HFESGB was found to be a generally acceptable source of HF guidance for the design of the System 80+ HSI.

18.7 Procedures

The HFE PRM for advanced evolutionary reactors specified that a review of Plant and Emergency Operating Procedure Development (Element 7) should be performed. The staff's DSER review of the CESSAR-DC identified

DSER Issues 18.9.1 - Operating Support Information Program (OSIP) and 18.9.2 - Emergency Operations Guidelines, and 20.2-3 (Issue I.C.1, ABB-CE committed to modify EOGs to ensure compatibility with System 80+ design). This review addresses these DSER issues.

18.7.1 Objectives

The objective of this review is to ensure that HFE principles and criteria are applied along with all other design requirements to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, and validated. The types of procedures covered in this element are

- plant and system operations (including startup, power, and shutdown operations)
- abnormal and emergency operations
- preoperational, startup, and surveillance tests
- alarm response

18.7.2 Methodology

18.7.2.1 Material Reviewed

The following ABB-CE documents were used in this review:

- Reference 5 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 5, "CESSAR-DC Markups for V&V Procedures," ABB-CE letter dated September 24, 1993.
- CESSAR-DC Section 18.9.3.2.

18.7.2.2 Review Scope

The staff initially envisioned that detailed POPs would be delivered as part of the System 80+ certified design, and as such considered that the OSIP would serve as the avenue for the development of such POPs. Based on further review of the application and the requirements of 10 CFR Part 52, the staff determined that development of POPs was in fact beyond the scope of the System 80+ design certification and will be the responsibility of the COL applicant. However, as described in the HFE PRM - Element 7 - Procedures - certain vendor technical documentation was required to support the COL's POP development process. The staff, therefore, redirected its efforts to review ABB-CE's program to assure that the vendor's technical information important for POP development was incorporated into the material provided to the COL applicant as part of the certified design.

18.7.3 Results

18.7.3.1 DSER Issues Review

18.7.3.1.1 DSER Issues

18.7.3.1.1.1 DSER Issue 18.9.1 - Operational Support Information Program (OSIP)

In the DSER, the staff stated that ABB-CE should provide the following information concerning the contents of the OSIP:

(a) the OSIP development process including,

(1) the scope of the OSIP (e.g., types of procedures and training covered, types of guidance documents such as procedure writer's guides, and verification and validation guides covered)

(2) the basis of the OSIP development (e.g., consideration of plant design basis, function, and task analysis; PRA/HRA-identified human actions)

(b) indicate how OSIP is expected to integrate with the results of the Nuclear Power Oversight Committee (NPOC) Block 7, "Enhanced Standardization Beyond Design," activities.

18.7.3.1.1.2 DSER Issue 18.9.2 - Emergency Operations Guidelines

At the time of the System 80+ DSER development, ABB-CE had not provided the staff copies of the System 80+ EOGs for staff review. In the DSER, the staff requested that ABB-CE submit information including (1) System 80+ EOGs, and (2) an analysis of the differences between the System 80+ EOGs and the current NRC-approved CEN-152, Revision 3, EOGs. The staff also requested that ABB-CE identify the differences between the current ABB-CE Owners Group EOGs and the System 80+ EOGs and the bases for the differences for each step of the EOGs.

18.7.3.1.1.3 DSER Issue 20.2-3 - ABB-CE Commitment to Modify EOGs

ABB-CE committed to modify the EOGs within current CEN-152 structure to ensure operational compatibility with the System 80+ design and to include an appropriate analytical basis.

18.7.3.1.2 Issue Resolution

18.7.3.1.2.1 DSER Issue 18.9.1 - Operational Support Information Program (OSIP)

Issue: The staff indicated that ABB-CE should provide the information identified above concerning the contents of the OSIP.

Evaluation: The information ABB-CE described in CESSAR-DC (e.g., Section 13.5.1, "Plant Operating Procedures Development Plan," and Section 18.9.3.2, "Operating Ensemble Validation Plan") includes, but is not limited to (1) the detailed task analysis, (2) complete event scenarios, data, results, and acceptance criteria from the design validation exercises, (3) applicable procedure development guidelines (e.g., emergency operating procedures guidelines), and (4) additional plant design basis material including the results of the PRA effort.

The staff reviewed the applicable CESSAR-DC sections against the requirements outlined in the HFE PRM to ensure that the important vendor's technical information required for the COL applicant's POP development process was identified. The staff concluded that ABB-CE had adequately described the required technical information. Therefore, Items a.1 and a.2, above, are resolved.

In response to the staff's RAI concerning the NPOC/OSIP integration, ABB-CE described that the program was under initial development and that detailed programmatic processes to ensure integration of the OSIP efforts with the NPOC work had not been fully developed. As a result of the staff's determination that the development of POPs will be a COL applicant activity, the staff concluded that it was premature to require programmatic details associated with the NPOC activities during the certified design application review. The staff will, therefore, review this aspect of POP development as part of the review of a COL application. Therefore, Item b, above, is resolved. DSER Issue 18.9.1, Operational Support Information Program (OSIP) is resolved.

18.7.3.1.2.2 DSER Issue 18.9.2 and DSER Issue 20.2.3 - Emergency Operations Guidelines and ABB-CE Commitment to Modify EOGs

ABB-CE submitted the System 80+ EOGs for the staff's review. The EOGs are a revision to the latest version of ABB-CE's report, CEN-152, on emergency procedure guidelines for its operating plants and reflect the design features of System 80+. CEN-152 has been reviewed and approved by the staff.

ABB-CE also submitted a deviation document identifying the procedural differences from CEN-152 along with supplemental information to explain the technical bases for the deviations. During the staff's review, ABB-CE provided responses, in a letter dated September 1, 1993, to staff review comments included in RAIs Q440.223 through Q440.246.

The staff reviewed the EOGs for System 80+, the deviation document, and the responses to RAIs. The staff concludes that the System 80+ EOGs are adequate and acceptable. The staff's acceptance of the EOGs is based on the following:

(1) The EOGs retain the structure and event mitigation strategies of CEN-152. The EOGs contain both symptom-oriented and function-based procedure guidelines. The symptom procedure guidelines include the procedure guidance for standard post-trip actions, reactor trip recovery, excess steam demand, loss-of-coolant accident, loss of offsite ac power, total loss of feedwater, steam generator tube rupture, and station blackout. The function recovery guidelines (FRG) address the safety functions such as reactivity control, maintenance of vital power sources, reactor inventory and pressure control, RCS and core heat removal, containment temperature and pressure control, containment isolation, and containment combustible gas control.

(2) The EOGs have been modified to reflect the System 80+ design including design features such as: four SI pumps (instead of two high pressure and two low pressure SI pumps in ABB-CE's existing plants), additional emergency feedwater pumps, interchangeability of containment spray and shutdown cooling pumps, in-containment refueling water storage tanks, alternate ac power supply, and safety depressurization system.

(3) The EOGs adequately incorporate the procedure guidelines required for the closure of DSER Open Items. The EOG changes for resolution of items are:

   (a) SI flow rate at the low pressure range - see Section 6.3.1 of this report for the resolution of DSER Open Item 6.3.1-1.

   (b) Use of the RCGS for RCS pressure control - see Section 6.7.1 of this report for the resolution of DSER Open Item 6.7.1-1.

(c) Use of the RDS for the feed-and-bleed operation - see Section 6.7.2 of this report for the resolution of DSER Open Item 6.7.2-4.

(d) Procedure changes reducing challenge to the primary safety valves to open during an SGTR event - see Section 15.3.9 of this report for resolution of DSER Open Item 15.3.8-1.

(e) Avoidance of de-boration during an SGTR event - see Section 15.3.9 of this report for resolution of DSER Open Item 15.3.8-2.

(f) Use of a dedicated seal injection system for RCP seal cooling - see Section 20, Item GSI-023 of this report for resolution of DSER Open Item 20.2-7.

(4) The most significant change to the containment isolation FRG was the design specific value for verifying containment isolation. This value of 19 kPa (2.7 psig) is consistent with the high containment pressure nominal trip setpoint given in Table 7.2-4 of CESSAR-DC. For the containment temperature and pressure control FRG, the most significant changes were the design specific values for verifying actuation of containment spray and the eventual termination of containment spray. The EOGs direct the operator to verify containment spray actuation when containment pressure reaches 59 kPa (8.5 psig) and to terminate containment spray once containment pressure is reduced below 38 kPa (5.5 psig). Table 7.3-5 of CESSAR-DC lists the containment spray actuation signal as 59 kPa (8.5 psig). Containment sprays are terminated at 38 kPa (5.5 psig) to reduce the possibility of wetting electrical connectors which may result in electrical grounds. There were no significant changes to the containment combustible gas control FRG. The staff believes that the differences, as provided by ABB-CE, between the System 80+ EOGs and the EPGs contained within CEN-152 do not affect the structure and event mitigation strategies of the previously approved guidance and are therefore acceptable.

(5) Another major difference between CEN-152 and the System 80+ EOGs is the addition of Appendix A, "Severe Accident Management Guidance," to the System 80+ EOGs. This appendix provides guidance on when to actuate the mitigative features that have been incorporated into the System 80+ design in order to cope with the consequences associated with a severe accident. Guidance has been provided for the following systems: safety depressurization, cavity flooding, hydrogen igniters, external connection for internal containment spray, and containment venting. The safety depressurization system is to be actuated by the operator when a primary safety valve lifts. The cavity flooding system and the hydrogen igniters are to be actuated upon diagnosis of a severe accident condition and when core exit temperatures exceed 371 °C (700 °F). Appendix A stipulates that a severe accident can be diagnosed based on the unavailability of the safety injection system and a low and continuously decreasing reactor coolant level as indicated by the reactor vessel level monitoring system. Guidance on the use of the external connection for internal containment spray and the containment vent is to be given to the operator by the technical support center. The staff finds this guidance adequate to ensure that appropriate decisions can be made by the plant operator during a severe accident until the technical support center can be established.

Since ABB-CE provided adequate EOGs for System 80+, the staff concludes that DSER Open Item 18.9.2-1 and DSER Confirmatory Item 20.2-3 are resolved.

18.7.3.2 HFE PRM Criteria-Based Evaluation

As a result of the staff's determination that the development of plant operating procedures will be a COL applicant activity, HFE PRM criteria 1 and 2 were considered to be relevant to the present concern of System 80+ design certification. The focus of the HFE PRM criteria-based evaluation was on the ABB-CE EPGs, which provide important input to the procedures of a COL applicant.

(1) Criterion 1: The task analysis shall be used to specify the procedures for operations (normal, abnormal, and emergency), test, maintenance, and inspection.

Evaluation: ABB-CE's FTA methodology uses the EPGs to evaluate operator task requirements for specific scenarios. Results of the task analyses are incorporated into the EPGs. In addition, ABB-CE has made a commitment to make available to a COL applicant the task analysis results that are relevant to training and procedures (see FSER Section 18.5.3.1.2). Hence, this criterion is satisfied.

(2) Criterion 2: The basis for procedure development shall include

- plant design bases
- system-based technical requirements and specifications

- task analyses for operations (normal, abnormal, and emergency)
- significant human actions identified in the HRA/PRA
- initiating events to be considered in the EOPs shall include those events present in the design bases

Evaluation: The information discussed with regard to DSER Issue 18.9.1 above satisfies this criterion.

18.7.4 Procedures Findings

The staff found that the ABB-CE approach to System 80+ procedure development was acceptable and that the information that will be provided by ABB-CE to the COL applicant is satisfactory to support the development of plant operating procedures.

18.8 Verification and Validation

The NRC HFE PRM for advanced evolutionary reactors specified that a formal V&V (Element 8) of the HSI should be performed. The staff's DSER review of the CESSAR-DC has identified a DSER issue related to HFE PRM Element 8 (i.e., DSER Issue 18.10-1).

18.8.1 Objectives

The objective of this review is to provide comments on the ABB-CE plan related to HFE PRM Element 8 - Verification and Validation.

18.8.2 Methodology

18.8.2.1 Material Reviewed

The following ABB-CE documents were used in this review:

- Reference 15 of CESSAR-DC Section 18.10, LD-93-071, "System 80+ Submittal #1 Design Descriptions and ITAAC,". ABB-CE letter dated April 30, 1993.
- Reference 16 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 2, "Justifications of ABB Positions Requested for Closure of V&V;" and Attachment 5, "SSAR-DC Markups for V&V and Procedures" ABB-CE letter dated September 24, 1993.
- Reference 3 of CESSAR-DC Section 18.7, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992.
- Reference 3 of CESSAR-DC Section 18.4, "Human Factors Engineering Verification and Validation Plan for Nuplex 80+" (NPX80-IC-VP790-03, Rev. 00, September 24, 1993), hereafter referred to as the plan.
- Reference 4 of CESSAR-DC Section 18.4, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 02, September 29, 1993), hereafter referred to as HFPP.
- CESSAR-DC, Sections 13.5 and 18.9.

18.8.2.2 Review Scope

The scope of this review was centered on the V&V plan, although additional ABB-CE documents were consulted (as referenced above).

The review focused on (1) resolution of DSER issues, and (2) evaluation of the ABB-CE documents with respect to the topics and general criteria of the HFE PRM. Complete adherence to the HFE PRM was not considered to be mandatory. Differences in approach would be considered acceptable provided (1) the program can still meet the HFE commitment and goals, (2) the difference between the proposed criteria and those contained in the HFE PRM are adequately justified, and (3) there is no adverse impact on other program elements.

18.8.2.3 Review Procedure

As indicated above, the staff's DSER review of the CESSAR-DC identified a DSER issue related to HFE PRM Element 8 (i.e., DSER Issue 18.10-1). The draft plan was developed following a public meeting held on September 10 and 11, 1992, between the staff and ABB-CE to address DSER issues. The plan was reviewed using the HFE PRM Element 8 general criteria as they would apply to an implementation plan (as contrasted to a final report of the V&V effort). The focus of an implementation plan is to provide the methodology by which the general criteria of the HFE PRM element are to be accomplished. Thus, the Plan was evaluated in terms of the HFE PRM general criteria and the methodology proposed for the V&V activities.

The following materials were consulted as part of the evaluation:

- American National Standards Institute, ANSI/ANS 3.5," Nuclear Power Plant Simulators for Use in Operator Training," Santa Monica, California, 1985.

- NRC HFE forwarded to the Commission in SECY-92-299, dated August 27, 1992.

- NRC Internal Memorandum, "Review of ABB-CE I&C Diversity Analysis Regarding Operator Response Times," June 25, 1993.

- Public meeting minutes from September 10 and 11, 1992, meetings between NRC and ABB-CE.

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Human Factors Program Plan for the System 80+ Standard Plant Design," (NPX80-IC-DP790-01, Rev. 01, December 15, 1992), ABB-CE letter dated December 18, 1992.

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 1, "Human Factors Engineering Verification and Validation Plan for Nuplex 80+," (NPX80-IC-VP790-03, draft plan), ABB-CE letter dated January 18, 1993.

- U.S. Nuclear Regulatory Commission (1992), "Draft Safety Evaluation Report" (NUREG-1492), Washington, D.C.

18.8.3 Results

18.8.3.1 DSER Review

18.8.3.1.1 DSER Issue

In the staff's initial review of this element, DSER Issue 18.10-1 was identified with concerns including

- establishment of V&V criteria to support the assessment of test results

- incorporation of "human-centered" operator performance such as operator workload in V&V tests

- verifying that the integrated CR supports the staffing requirements of 10 CFR 50.54(m)

At the September 10 and 11, 1992, public meeting, ABB-CE agreed to address these concerns in a V&V plan to include the following 15 items

(a) identification of a schedule for a validation report

(b) evaluation of design goals and functional requirements

(c) criterion 2 of the HFE PRM

(d) evaluation of availability and suitability of HSI elements

(e) evaluation of integration of HSI elements with each other and personnel including HSI prototypes and plant simulator

(f) the dynamic evaluations in Criterion 5 of the HFE PRM

(g) the performance measures for dynamic evaluations included in Criterion 6 of NRC's HFE program review model

(h) verification that all issues addressed in ABB-CE's HFE issues tracking system have been addressed

(i) verification that critical human actions have been supported in the design

(j) operational definitions of "adequate" and "acceptable"

(k) demonstration that CR design accommodates the staffing requirements of 10 CFR Part 50.54(m)

(l) specification of additional skill areas required other than HFE specialists and operations experts to perform a formal analysis

(m) evaluation of operator aids

(n) demonstration of acceptable operator performance

(o) evaluation of operator performance under degraded conditions including complete failure of the DPS

18.8.3.1.2 Issue Resolution

These 15 items addressing the DSER issue are described in Section 18.8.3.2. Table 18.6 provides a cross-reference between the item specification and the appropriate V&V plan section and subsection of 18.8.3.2 in this document where the item is addressed.

18.8.3.2 HFE PRM Criteria-Based Evaluation

According to HFE PRM General Criterion 1, the V&V evaluation ensures that the performance of the HSI, when all elements are fully integrated into a system, meets (1) all HFE design goals as established in the program plan, and (2) all system functional requirements and support human operations, maintenance, test, and inspection task accom

Table 18.6 Resolution of V&V DSER issue items

ITEM	PLAN SECTION	SER SECTION 18.8.3
a	7	2.7
b	6.3	2.6
c	6.1.2, 6.2.2, 6.3.2	2.1
d	6.1, 6.2	2.4, 2.5
e	6.3	2.6
f	6.3.4.2	2.6
g	6.3.4.1, 6.3.5	2.6
h	not addressed	2.3
i	6.3	2.6
j	(see Note 1)	2.6
k	6.3.5	2.6
l	6.1, 6.2, 6.3	2.3, 2.4, 2.5, 2.6
m	not addressed	(see Note 2)
n	6.3.5	2.6
o	6.3.4.2	2.6

Note 1. Operational definitions of "adequate" and "acceptable" are not specifically addressed: However, the V&V methods provide criteria that address what constitutes adequacy and acceptability.

Note 2. Operator aids are not specifically addressed, however, the identification of aids should be accomplished through the V&V analyses.

plishments. This is done through a set of evaluations which are described below.

18.8.3.2.1 Scope

Criterion: HFE PRM General Criterion 2 states that V&V evaluations shall address

- human-hardware interfaces
- human-software interfaces
- procedures
- workstation and console configurations
- control room design
- remote shutdown system
- design of the overall work environment

Evaluation: The plan scope is identified in several areas. A general scope statement appears in Section 2 and 6.2.4 which identifies the scope as all HSIs in the MCR, remote shutdown area (RSA), and the LCSs specified in the EOGs. Procedures are specifically excluded. The staff interpret the reference to HSIs to include all of the above with the two exceptions discussed below.

First, in a draft version of the plan, it was not clear whether V&V activities would be directed toward environmental considerations such as lighting and noise in the MCR and lighting, noise, temperature, etc. at local panels. Following discussions with the staff, ABB-CE incorporated "workplace environment" into the defined scope in Section 2 of the plan. On the basis of this plan revision, this issue is resolved.

Second, the issue of the absence of a procedure element from the HFE program has already been identified and resolved in FSER Section 18.2 - Human Factors Engineering Program Management. One of the staff's concerns regarding exclusion of procedures from ABB-CE's HF program, was its impact on validation. HFE PRM Element 8 validation includes final procedures and their interactions with the rest of the HSI as an essential aspect of HFE validation. As part of the resolution to the general procedure issue, ABB-CE included as part of its CESSAR-DC a requirement in COL Action Item 13.5.1, Plant Operating Procedure Development Plan, for the COL to perform a POP validation effort that demonstrates the acceptability of the completed procedures. CESSAR-DC Section 18.9.3, Validation, was then modified to break validation into two phases. Section 18.9.3.1, Design Validation, addresses validation of the entire HSI without final procedures. CESSAR-DC Section 18.9.3.2, Operating Ensemble Validation Plan, and Section 6.3.4.4 of the plan, Operating Ensemble Validation Activities, addresses the "final" validation of the HSI after the final procedures have been completed. Operating ensemble validation requirements are addressed in CESSAR-DC Section 19.3.1.2. This validation, which will be performed by the COL, will provide assurance that trained operators using final, plant-specific procedures in the as-built CR form an effective operating ensemble. This two-phased validation approach is acceptable to the staff, since together they meet the scope requirements of the HFE PRM. Based upon the establishment of the approach to validation described in CESSAR-DC Sections 18.9.3 and 15.5, this criterion is satisfied.

18.8.3.2.2 Technical Basis in Current Literature

Criterion: HFE PRM General Criterion 9 states that the V&V effort shall be performed using the set of identified documents as guidance (see the HFE PRM for the specific list). The purpose of this criterion is to ensure that the HSI is "evaluated using accepted HFE principles based upon current HFE practices."

Evaluation: Sections 3 and 18.4 of the draft plan identified documents as V&V plan references. Many of these documents correspond to the documents identified in HFE PRM General Criterion 9. However, while the section is entitled references, most of the documents listed were not specifically referenced nor is it clear how they were used. For example, EPRI NP-3701, Computer-Generated Display System Guidelines, was referenced; yet in the section on suitability verification (where the document would most likely be applied), it was not identified as a criteria document. An examination of the verification analysis report did not indicate that anything other than NUREG-0700 was used for verification (which does not contain sufficient criteria verification of a CR such as the System 80+). The revision to the plan provided the specific references to the HFE PRM recommended technical basis documents. The verification analysis will be based upon criteria from a broad basis of HFE PRM identified documents (including, but not limited to, NUREG-0700) and additional acceptable industry sources. Four HFE PRM identified sources were noted (AR 602-1, TOP 1-2-610, DODI 5000.2, and EPRI NP-3701). These exceptions were acceptable to the staff since their contribution to the ABB-CE V&V effort was redundant with the documents cited. (The review and acceptance of the specific criteria used for verification is addressed in FSER Section 18.6.3, HFE Standards, Guidelines, and Bases.) The staff, therefore, determined that the technical basis of the ABB-CE V&V plan was acceptable. Based upon the revisions to the plan, this criterion is satisfied.

18.8.3.2.3 Human Factors Issue Resolution Verification

Criterion: HFE PRM Criterion 7 states that a verification shall be made that all issues documented in the HF issue tracking system have been addressed.

Evaluation: The staff noted that verification of HFE issues resolution was not addressed in the draft plan. Following discussions with the staff, ABB-CE addressed the concern in two ways. First, assurance of closeout of tracking system items is incorporated into the description of the tracking of issues description of the HFPP. Second, Sections 6.1.4 (Availability Verification), 6.2.2 (Suitability Verification), and 6.3.4.1 (Validation) of the plan were modified to require that relevant TOI items are addressed in the appropriate V&V activities which are required as part of ITAAC and DAC. The staff finds this an acceptable approach to ensure HFE issue resolution verification. Based upon the revisions to the V&V plan and the HFPP, this criterion is satisfied.

18.8.3.2.4 HSI Task Support (Availability) Verification

Criterion: HFE PRM Criterion 3 states that individual HSI elements shall be evaluated in a static and/or "part-task" mode to assure that all controls, displays, and data processing that are required to accomplish human safety-related tasks and actions [as defined by the task analysis, EOP analysis, and PRA/human reliability analysis (HRA)] are available through the HSI.

Evaluation: Plan Section 6.1 describes the approach to availability verification. ABB-CE's availability analysis accomplishes two objectives: first, consistent with HFE PRM Criterion 3 to ensure that all required HSI elements are available; second, to identify HSI elements that are not required for task accomplishment so that they can either be removed or relocated to the appropriate place. The latter objective is consistent with one of the purposes of HFE verification in the HFE PRM and is described in draft NUREG/CR-5908.

In the draft plan, the scope of the tasks to be analyzed was unclear. Under "Purpose," item one identified "operator tasks" with no qualification. Under purpose of availability analysis, the "Procedure Guideline Information & Control Requirements" (PGICR) was identified along with the minimum inventory and federally mandated requirements. The PGICR was defined as "a summarization of procedure-based parametric requirements." The scope of Phase 1 was then tied to those aspects of the HSI that are "specified in the EOG." It was unclear whether the availability analysis would be limited to EOG-based actions, since PRA critical tasks, normal operations, and abnormal operations should be addressed as well. ABB-CE agreed and modified Section 6.1 of the plan to clearly incorporate tasks included in the staff's concern. Based upon the revisions to the plan, this criterion is satisfied.

Criterion: Implementation Plan Requirements for Methodology Specification (relevant to this verification)

- general objectives
- methodology and procedures
- participants
- analysis
- criteria for evaluation of results
- utilization of evaluations

Evaluation: While most of the information identified above was acceptably provided in the draft plan, several concerns regarding methodology were identified by the staff:

- In the draft plan, the availability analysis criteria for SPDS did not include NUREG-1342 and the post-accident monitoring indications did not include RG 1.97. ABB-CE incorporated these criteria in Section 6.1.5.1 of the plan, Items a and l.

- In the draft plan, the availability analysis criteria included the term "System I&C Inventory" only. The staff did not consider this a criterion so ABB-CE revised the criteria in Section 6.1.5, Item 4, to indicate that each "System I&C Inventory" entry has a specified basis.

- The Phase 2 methodology addresses identification of HSI elements that are not required for task accomplishment. The draft plan, stated that an "unnecessary" aspect of the HSI would be anything not on the list resulting from availability analysis as defined and would require removal of anything not on the list (note: "The process will be repeated until the HSI panel designs match the availability checklist"). The staff was concerned that the possibility existed that any incompleteness or problems with list generation could result in the removal of operationally significant information. The staff recommended that an operational review of candidate items for removal be performed to assure that no information important to plant operations be removed. ABB-CE has committed to perform an operational review of any information identified as unnecessary to confirm that no information of operational significance is lost. Section 8.1 of the plan was revised accordingly.

Since the staff concerns were all addressed in the revised plan, these criteria are satisfied.

18.8.3.2.5 HFE (Suitability) Verification

Criterion: HFE PRM Criterion 3 states that individual HSI elements shall be evaluated in a static and/or "part-task" mode to assure that all controls, displays, and data processing that are required are designed according to accepted HFE guidelines, standards, and principles.

Evaluation: Suitability verification is addressed in Section 6.2 of the plan. The stated purpose of suitability verification is consistent with the HFE PRM criterion. The applicant's suitability evaluations will compare the HSIs with accepted HFE guidelines, standards, and principles. HSI prototypes and mock-ups will be used to perform the evaluations, thus providing static or dynamic representations as required by the specific area being evaluated, therefore, this criterion is satisfied.

Criterion: Implementation Plan Requirements for Methodology Specification (relevant to this verification)

- general objectives
- methodology and procedures
- participants
- analysis
- criteria for evaluation of results
- utilization of evaluations

Evaluation: The draft plan generally addresses the above requirements. While the staff considered ABB-CE's approach to HFE (suitability) verification to be generally good, several concerns were raised.

(1) Suitability is addressed using both a top-down and bottom-up approach. The top-down approach addresses the appropriateness of design selections within the context of operator tasks (which is consistent with "appropriate use" considerations in draft NUREG/CR-5908). However, the methodolo-

gy Section, 6.2.4, mainly addressed the bottom-up approach. Consultation of the verification analysis report appeared to have the same limitation. ABB-CE expanded Section 6.2.4 of the plan Methodology to elaborate their approach to suitability analysis. The top-down approach is clarified in Sections 6.2.4 and 6.2.4.1. The approach evaluates the HSI in terms of usability with respect to the anticipated task demands. This is an appropriate aspect to suitability analysis especially for newer features of an advanced CR for which guidance based upon historical technology applications is limited. The staff, therefore, finds this approach acceptable.

(2) The bottom-up approach uses HFE guidance as a basis. In the draft plan, the criteria identified were limited to NUREG-0700 and ABB-CE's HFE standards, guidelines, and bases for System 80+ (NPX-IC-DR-791-02). It was unclear whether these were to be the only criteria or whether additional documents (such as those identified in Section 3 - References) would be utilized. As indicated in FSER Section 18.8.3.2.2 above, the revision to the plan provided the specific references to the HFE PRM recommended technical basis documents. The verification analysis will be based upon criteria from a broad basis of HFE PRM identified documents (including, but not limited to, NUREG-0700) and additional acceptable industry sources. (The review and acceptance of the specific criteria used for verification is addressed in FSER Section 18.6.3 HFE Standards, Guidelines, and Bases.) The staff, therefore, determined that the technical basis of the ABB-CE V&V plan was acceptable.

(3) In the draft plan, it was unclear whether all elements in the HSI (e.g., every display) would be reviewed or whether a sampling process would be used. ABB-CE modified Section 6.2.4.1 of the plan to indicate that all elements of the HSI would be reviewed rather than using a sampling process. This approach is acceptable to the staff.

(4) The draft plan did not indicate how discrepancies from guidance checklists would be resolved. It is the staff's position that conformance to any specific individual guideline should not, in itself, be a requirement because guidelines are insensitive to the trade-offs between design features and functions that typically occur in final designs. These trade-offs may result in discrepancies between an acceptable final design and a specific guideline. Instead a verification against generic guidelines should identify potential concerns which should be addressed, but which may be perfectly acceptable due to a technical basis in design studies, tests, and trade-off analyses as justified by the designer. Following discussions with the staff, ABB-CE elaborated the treatment of discrepancies. Discrepancies will be treated as potential concerns until examined by the applicable review process and resolved by the responsible management structure. This review process is described in Sections 5.4 and 8.1 of the plan. This approach is consistent with the staff's position and is acceptable.

Since the staff concerns were all addressed in the revised plan, these criteria are satisfied.

18.8.3.2.6 Integrated System Validation

Criterion: HFE PRM Criterion 4 states that the integration of HSI elements with each other and with personnel shall be evaluated and validated through dynamic task performance evaluation using evaluation tools which are appropriate to the accomplishment of this objective. A fully functional HSI prototype and plant simulator shall be used as part of these evaluations. If an alternative to an HSI prototype is proposed its acceptability shall be documented in the implementation plan. The evaluations shall have as their objectives to confirm the

- adequacy of entire HSI configuration for achievement of safety goals
- allocation of function and the structure of tasks assigned to personnel
- adequacy of staffing and the HSI to support staff to accomplish their tasks
- adequacy of Procedures
- adequacy of the dynamic aspects of all interfaces for task accomplishment
- evaluation and demonstration of error tolerance to human and system failures

Evaluation: Validation is discussed in Section 6.3 of the plan. The staff identified several concerns regarding validation in the draft plan.

(1) The HFPP and the CESSAR-DC (Sections 18.9.2 and 18.9.3) referred to phased validation, but the validation description in the plan makes no such distinction. ABB-CE revised the descriptions in all the above documents to make them consistent with the phased validation approach (as discussed in

FSER Section 18.8.3.2.1, Scope, above). This was acceptable to the staff.

(2) The draft plan did not clearly indicate that a dynamic plant simulator will or will not be used. The staff also noted that the descriptions of validation test bed requirements were different in CESSAR-DC, the plan, and the draft ITAAC. Following discussions with the staff, an acceptable description was developed - "a facility that physically represents the MCR configuration [RSR configuration] and dynamically represents the operating characteristics and responses of the System 80+ design." This description has been uniformly incorporated into the above referenced documents. It was agreed that the testbed requirements for mockup of local control stations should be handled on a case-by-case basis as the detailed requirements for specific V&V activities become defined following certification.

(3) The purposes of the validation identified in the draft plan were consistent with the HFE PRM. However, the following concerns were identified

- role of procedures
- evaluation and demonstration of error tolerance to human failures

The procedure issue has been resolved in general (see FSER Section 18.7) and for validation in particular (see FSER Section 18.8.3.2.1, Scope). However, Section 6.3.1 makes references that EOGs and validation should include other procedures as well, e.g., normal procedures, abnormal procedures, and alarm response procedures. ABB-CE agreed and revised Sections 6.3.1 and 6.3.2 of the plan to include the broader scope of operational conditions.

With respect to the evaluation and demonstration of error tolerance to human failures, ABB-CE agreed to evaluate human errors which occurred during the validation tests. However, the System 80+ design does not include the specific error monitoring and checking features which this HFE PRM criterion was intended to validate (i.e., the criterion is more appropriate to a more advanced plant design). Thus, the staff determined that for an evolutionary design, ABB-CE's approach to validation error analysis was acceptable.

Since the staff concerns were all addressed satisfactorily in the revised plan, these criteria are satisfied.

Criterion: HFE PRM Criterion 5 states that the dynamic evaluations shall evaluate HSI under a range of operational conditions and upsets, and shall include

- normal plant activities (e.g., startup, full power, and shutdown operations)

- instrument failures (e.g., safety system logic & control (SSLC) unit, fault tolerant controller (NSSS), local "Field Unit" for MUX system, MUX controller (BOP), break in MUX line)

- HSI equipment and processing failure (e.g., loss of VDUs, loss of data processing, loss of large overview display)

- transients (e.g., turbine trip, loss-of-offsite power, station blackout, loss of all FW, loss of service water, loss of power to selected buses/CR power supplies, and SRV transients)

- accidents (e.g., main steamline break, positive reactivity addition, control rod insertion at power, control rod ejection, ATWS, and various-sized LOCAs)

Evaluation: Draft plan Section 6.3.4.2 identified the scenarios identified for validation. A total of 22 situations were defined which generally covered the evaluation classifications defined by the HFE PRM criterion. Several concerns were identified in the review of scenarios:

(1) Section 6.3.4.2, Emergency Operations - The staff expected that all operations based on EOPs and procedures that are based on System 80+ FRGs would be included in validation. While EPG-related scenarios were addressed, the treatment of FRGs was incomplete. In discussion with the staff, ABB-CE stated that ATWS and ESDE scenarios will be used to address FRGs. This acceptably addressed the staff's concern.

(2) Section 6.3.4.2, Abnormal Operations - The staff expected that scenarios reflecting (1) selected RCP failures, e.g., loss of seal cooling and injection, seal failure (a known PWR operational issue, GI-23); and (2) stuck open pressurizer relief valve (the TMI scenario) would be included in the validation tests. Following discussion with the staff, ABB-CE included these scenarios in Section 6.3.4.2 of the plan.

(3) Section 6.3.4.2, HSI and I&C Failure Sequences - Include scenarios reflecting (1) loss of selected instrument failures (e.g., L_{PZR}, T_H, T_C, etc.), and (2) Loss of IPSO in combination with emergency operations events/transients. Following discussion with the staff, ABB-CE agreed to consider the incorporation of the identified instrument failures when a detailed V&V implementation plan is

developed following certification and is submitted for staff review as per the HFPP. This commitment has been identified in Appendix B of the plan. In addition, one beyond-design-basis I&C failure scenario (common mode I&C failure during a LOCA incorporating a loss of DIAS-N, ESF-CCS, and PPS) has been added to Section 6.3.4.2 of the plan. (The scenario was added subsequent to the staff's review and documented in NRC internal memorandum dated June 25, 1993: ABB-CE's diversity analysis regarding operator response times, the staff concluded that ABB-CE provided a basis for operator response time estimates and that specific operator response time estimates are acceptable.) This approach is acceptable to the staff since scenario details are more appropriately addressed in the detailed implementation plan.

(4) PRA critical actions - The staff was concerned that not all PRA critical actions would be addressed in the defined scenarios. If not, construct scenarios to validate the accomplishment of these actions. Following discussion with the staff, ABB-CE agreed and modified Section 6.3.5 of the plan to assure that all PRA critical actions are addressed in validation operational sequences.

(5) The staff was concerned that the system would not be validated for tolerance to human error. This issue is discussed above with respect to HFE PRM Criterion 4 (the first criterion discussed in this section) and the staff agreed with ABB-CE's recommended approach to error evaluation.

Since the staff concerns were all addressed in the revised plan, these criteria are satisfied.

Criterion: HFE PRM Criterion 6 states that performance measures for dynamic evaluations shall be adequate to test the achievement of all objectives, design goals, and performance requirements and shall include at a minimum

- system performance measures relevant to safety
- crew primary task performance (e.g., task times, procedure violations)
- crew errors
- situation awareness
- workload
- crew communications and coordination
- anthropometry evaluations
- physical positioning and interactions

Evaluation: The staff was concerned that the draft plan did not specifically identify the data to be collected or how it would be analyzed. For example, no mention of system performance or task times was made. Since one of the stated objectives of the tests is "validation of time response for credited operator actions," it was expected that time would be measured. Section 6.3.4.1 of the plan generally discussed the collection of information related to a verbal protocol of operator actions and selected link analysis type data, but no clear presentation of data to be collected is presented. CESSAR-DC (Sections 18.9.2 and 18.9.3) discusses validation of anthropometrics and the assurance of adequate perceptual and cognitive load, yet these are not addressed by the draft plan.

Following discussion with the staff, ABB-CE agreed to address the staff's concerns in two ways. First, Section 6.3.4.3 of the plan was modified to define how data collection would be accomplished (via data event logging and subjective evaluation). Second, more detail regarding data specifications will be provided in a detailed V&V implementation plan developed following certification and is submitted for staff review as per the HFPP. This commitment has been identified in Appendix B of the plan. This approach is acceptable to the staff since validation test details are more appropriately addressed in the detailed implementation plan. Hence, these criteria are satisfied.

Criterion: HFE PRM Criterion 8 states that a verification shall be made that all critical human actions as defined by the task analysis and PRA/HRA have been adequately supported in the design. The design of tests and evaluations to be performed as part of HFE V&V activities shall specifically examine these actions.

Evaluation: See discussion under HFE PRM Criterion 5, Item 4 above regarding PRA critical actions. This criteria is satisfied.

Criterion: Implementation Plan Requirements for Methodology Specification (relevant to validation). A cross-reference is provided to the appropriate location where each item is discussed:

- general objectives (HFE PRM criterion 4 above)
- test methodology and procedures (see 1 and 2 below)
- test participants (see 3 below)
- test conditions (HFE PRM criterion 5 above)
- HSI description (see 4 below)
- performance measures and analysis (HFE PRM criterion 6 above)
- criteria for evaluation of results (see 5 below)

- utilization of evaluations (see 6 below)
- documentation (see 7 below)

Evaluation: The HFE PRM general criteria address most of the significant methodological considerations. The staff's review identified a number of concerns.

(1) In general, the methodology is described at a very general level for the purposes of a validation plan. The HFE PRM intention was for a plan which described the details of the validation effort. ABB-CE agreed to provide a more detailed V&V implementation plan to be developed following certification that will be submitted for staff review as per the HFPP. This approach is acceptable to the staff since validation test details are more appropriately addressed in the detailed implementation plan which can best be developed when the design becomes completed.

(2) The staff was concerned that use of "walkthrough" methodology implied that the underlying plant dynamics would be absent from validation. The absence of plant dynamics in validation would be inconsistent with the staff's position regarding validation tests. ABB-CE revised the reference in Section 6.3.4.1 of the plan to the use of the walkthrough methodology as a supplemental technique. Staff concerns regarding the testbed were addressed in the discussion of HFE PRM Criterion 4 above.

(3) The plan stated that the participants will be "operations experts" who are defined as "currently or formerly licensed reactor operators . . ." The staff was concerned that a more detailed description of the number of test participants and their qualification was needed for validation. ABB-CE agreed to provide a more detailed V&V implementation plan to be developed following certification that will be submitted for staff review as per the HFPP. Test participant requirements have been identified as a required item in the implementation plan appendix (Appendix B) to the plan. This approach is acceptable to the staff since validation test details are more appropriately addressed in the detailed implementation plan which can best be developed when the design becomes completed.

(4) The HSI was defined as a dynamic mockup of the MCR consoles that simulates plant operational responses. The staff was concerned about the ways in which the HSI might differ from the final design (in terms of the HFE significant aspects, e.g., response times, COL-selected equipment representations in displays, site-specific HSI characteristics). Following discussions with the staff, ABB-CE agreed that the facilities will meet the applicable portions of the requirements in Sections 4.1 and 4.2 of ANSI 3.5. This position will achieve an acceptable facility testbed. Staff concerns regarding the testbed were also addressed in the discussion of HFE PRM Criterion 4 above.

(5) With regard to criteria in Section 6.3.5 of the draft plan, the staff review identified the following concerns:

   (a) Operator errors made during scenarios should be examined to assess system response and tolerance.

   (b) The difference between the types of errors defined in 1a, 1b and 1g of the plan were unclear.

   (c) The criteria for addressing human error focused on very specific tasks, e.g., post-trip actions. Error evaluation should be accomplished for all operator tasks in the validation exercises.

   (d) As part of 2d, the criterion should include adequate work space for procedure usage.

   (e) Criterion 6c validates that operators can recognize an information or control failure within 15 minutes. A technical basis for 15 minutes was not provided.

With respect to a and c above, the staff has found ABB-CE's treatment of human error acceptable (see discussion under HFE PRM Criterion 4, Item 3 above). For b above, the description of errors in Section 6.3.5 of the plan has been revised to clarify the ambiguous terminology.

With respect to d above, ABB-CE has stated that laydown space will be evaluated as part of suitability verifications. Any issues related to laydown space that are identified during verification will be identified addressed through design modifications (as would any problem identified during V&V). This approach to the evaluation of laydown is acceptable.

With respect to (e) above, ABB-CE stated that the criterion was arbitrary and was removed from the plan. Concern for prompt response will be addressed through an evaluation of event logs and subjective evaluations. These details will be addressed in the more detailed implementation plan to be provided for staff review after certification.

(6) Utilization of results is discussed in Section 8 and is illustrated as part of the V&V process in Figures 7.1 to 7.3. Issue identification, resolution, and review are acceptably provided for in the process.

(7) Documentation is addressed in Section 7 of the plan. Each proposed V&V activity will generate a report that documents the activity and which will be reviewed by the design team (as per the HFPP). Each report is described in more detail in the appropriate analysis section.

Since the staff concerns were all addressed in the revised plan, these criteria are satisfied.

Additional Criteria: The staff review noted that there were issues raised during the review of other HFE PRM elements that were to be addressed in validation. These are identified below

(1) Task Analysis:

(a) interference of maintenance activities associated with I&C in the MCR and operations

(b) maintenance work order management and equipment tagout. ABB-CE has stated that most of this work will be done in the MCR, but outside the main control space. However, there is a requirement to interact with CR operators and this interaction should be evaluated

(c) equipment, documentation and supplies required to support personnel during normal, abnormal, and emergency operations. An important consideration is how CR personnel will use paper procedures in the CR. This includes considerations of task lighting, ease of handling, and adequacy of laydown. Similar evaluations should consider P&IDs, TSs, and other operator aids

(d) operator awareness of the status of equipment under surveillance test or repair

(2) HSI Design: It is the staff's position that the evaluation of the DPS and DIAS alarm implementation under high-alarm conditions should be specifically evaluated in validation.

Evaluation: These issues were not addressed in the draft plan. Following discussions with the staff, ABB-CE has addressed the issues in the following ways. With respect to Item 1a and b, Section 6.3.4.2 of the plan has been revised to include basic maintenance tasks during normal operations. With respect to Item 1c, the issues will be addressed during suitability analysis and any concerns observed during validation testing will be noted. With respect to Items 1d and 2, requirements for their evaluation will be included as part of the implementation in Appendix B of the plan.

Based upon these plan revisions, the criteria are satisfied.

18.8.3.2.7 Scheduling

Criterion: In the proposed resolution of the DSER, ABB-CE agreed to provide a schedule of V&V activities (as per the HFE PRM requirement in Element 1).

Evaluation: Scheduling is described in Section 7 of the plan. In Section 7.1 it states that availability verification can be accomplished "in parallel with, before, or after suitability verification." The staff was concerned that not all HSI changes resulting from availability verification would be subject to suitability verification. Following discussions with the staff, ABB-CE modified Section 6.2.2 of the plan to state that all HSI items will be verified as suitable. Based upon this plan revision, this criterion is satisfied.

18.8.4 Verification and Validation Findings

The ABB-CE approach to V&V has been reviewed and found acceptable. While the present plan is lacking complete methodological detail, a more detailed implementation plan will be developed following design certification. Requirements for the additional detail addressing staff concerns is provided in Appendix B of the plan. This approach is acceptable to the staff since V&V details are more appropriately addressed in a detailed implementation plan which can best be developed when the design becomes completed.

18.9 Certified Design Description/Inspections, Tests, Analyses, and Acceptance Criteria

18.9.1 Objectives

The objective of this review is to evaluate the System 80+ MCR ITAAC, remote shutdown room ITAAC, and control panels ITAAC against the requirements of 10 CFR Part 52.47(a)(1)(vi) and the HFE PRM.

18.9.2 Methodology

18.9.2.1 Material Reviewed

The following ABB-CE documents were used in this review:

- System 80+ ITAAC Section 12.2.1, "Main Control Room;" Section 2.12.2, "Remote Shutdown Room;" and Section 2.12.3, "Control Panels."
- Reference 15 of CESSAR-DC Section 18.10, LD-93-071, "System 80+ Submittal #1 Design Descriptions and ITAAC," ABB-CE letter dated April 30, 1993.
- Reference 13 of CESSAR-DC Section 18.10, LD-93-147, "System 80+ Information for Issue Closure," Attachment 1, "Response to Cross-Branch Chapter 18 Questions (October 4, 1991)," ABB-CE letter dated October 18, 1993.

18.9.2.2 Review Scope

The scope of this review was centered on the following System 80+ ITAAC and associated design descriptions: ITAAC Number 2.12.1, "Main Control Room;" ITAAC Number 2.12.2, "Remote Shutdown Room;" and ITAAC Number 2.12.3, "Control Panels."

The review focused on ensuring that significant features of the design certification application contained in the CESSAR-DC are captured by the CDD.

18.9.2.3 Review Procedure

As stated above, the staff's DSER review of the CESSAR-DC stated that ABB-CE must provide appropriate CR HF ITAAC, including DAC for portions of the design not completed at the time of the final design approval. Further, the staff noted that the ITAAC and DAC should be consistent with the criteria described in the HFE PRM.

By letter dated April 30, 1993 (Ref. 15 of CESSAR-DC Section 18.10, LD-93-071), ABB-CE submitted to the NRC for review and approval ITAAC and associated design descriptions for the MCR, remote shutdown room, and control panels.

The ITAAC and CDD were reviewed using the requirements of the HFE PRM and Part 52. Staff comments were discussed with ABB-CE at the public meeting held October 4 through 6, 1993. The resolution of staff comments is documented in minutes of that meeting.

The following materials were consulted as part of the evaluation:

- HFE PRM, forwarded to the Commission in SECY-92-299, dated August 27, 1992.
- Public meeting minutes from October 4 through 6, 1993, meeting between NRC and ABB-CE on ITAAC.
- U.S. Nuclear Regulatory Commission (1992), "Draft Safety Evaluation Report, (NUREG-1492), Washington, D.C.
- U.S. Nuclear Regulatory Commission (1993), "Form and Content for a Design Certification Rule" (SECY-92-287A), March 26, 1993, Washington, D.C.

18.9.3 Results

18.9.3.1 General Scope

The review of the CESSAR-DC (through Amendment Q) using the HFE PRM led to the staff's conclusion that the design and implementation process contained the necessary and sufficient aspect of an HFE program to result in an acceptable HSI design. The general guidance provided in SECY-92-287A was used to support the CDD ITAAC and DAC review. ABB-CE's CDD was compared to the major HFE PRM elements to determine whether they were captured. The following five elements were excluded from ITAAC: Human Factors Engineering Program Management, Operating Experience Review, System Functional Requirements Analysis, Allocation of Function, and Procedures Development. The first four exclusions were allowed because the staff had previously reviewed and found acceptable ABB-CE's documentation with respect to the first four of the identified HFE PRM elements. HFE PRM Element 7, Procedures Development, has been identified as a COL action item. This was found acceptable (see FSER Section 18.7). The remaining HFE PRM elements - Task Analysis, Human-System Interface Design, and Human Factors Verification and Validation - were addressed in the CDD and ITAAC and DAC.

At the October 4 through 6, 1993, public meeting, ABB-CE proposed to delete the control panels ITAAC. The basis for this proposal was that it was nearly wholly redundant with the MCR and RSR ITAACs. ABB-CE proposed acceptable disposition of the other control panel design commitments as follows. The control panel configuration figure will be added to CESSAR-DC Chapter 18. The control panel suitability is a part of the MCR and RSR suitability per design commitment 3 in the respective ITAACs. The task execution control panel design commitment will be in the MCR and RSR ITAACs (i.e., design commitment 4). In addition, task analysis output is an input to the availability inspection acceptance criteria of design commitment 2, as discussed in the System 80+ V&V plan. The design description statement in the control panels ITAAC regarding control panel seismic category was added to the MCR and RSR design descriptions. The staff found this approach and commitments acceptable.

18.9.3.2 Level of Detail

The ITAAC were evaluated to ensure that they accurately reflected the design and implementation process and that they were at a level of detail consistent with the staff's intent not to constrain the use of state-of-the-art, proven technology at the time the HSI is designed (one of the stated intents of the DAC concept). All necessary and sufficient ITAAC were identified based upon comparison to the HFE PRM and no concerns were identified.

Therefore, the staff concludes that the design commitments in the HFE ITAAC and DAC accurately summarize the Design Description for HFE; that the inspections, tests, and analyses identified are acceptable methods for determining whether the design commitments have been met; and that the acceptance criteria are sufficient to establish, if they are met, that the design commitments have been met.

18.9.3.3 Main Control Room Minimum Inventory

18.9.3.3.1 Discussion in the CESSAR-DC

ABB-CE's initial CESSAR-DC provided insufficient information about controls, displays and annunciators to be utilized for the System 80+ CR, resulting in a staff RAI. As part of the general resolution of the lack of CR detail, ABB-CE provided the detailed CR design implementation process, through which the specific controls, displays, and annunciators will be specified and designed. However, in order to provide an initial set of controls, displays, and annunciators for transient mitigation before design certification, ABB-CE developed the inventory described in CESSAR-DC Chapter 18. This inventory was developed by analyzing the System 80+ EPGs and the important operator actions specified as a result of the System 80+ PRA analysis. Subsequently, ABB-CE described an additional fixed-position subset of these controls, displays, and annunciators (i.e., a minimum inventory) for inclusion in the MCR ITAAC and design description (Table 2.12.1-1). This subset was also based on FTA data and results, which supported ABB-CE's identification of important displays, controls, and annunciators for EPG implementation.

ABB-CE submitted a letter (Ref. 13 of CESSAR-DC Section 18.10, LD-93-147), describing the technical basis for the MCR ITAAC minimum inventory which included (1) the criteria for inclusion of specific annunciators, displays, and controls, (2) the development method and scope of the minimum inventory, and (3) important operator actions identified through the PRA analysis including indications and controls required for each. The staff reviewed the supplemental material provided and determined that it satisfactorily addressed the staff's concerns with regard to development of the minimum inventory. The staff's review of the issue is complete, and the minimum inventory of displays, controls, and annunciators is considered adequate.

18.9.3.3.2 Analysis

18.9.3.3.2.1 Review Methodology

The staff reviewed the System 80+ FTA and supporting documentation to ensure that the inventory provides a reasonable minimum set of fixed controls, displays, and alarms to adequately implement the EPGs for the System 80+ design and account for the critical operator actions identified through the System 80+ PRA effort.

The analysis methods used for this evaluation included:

(1) <u>EOG Review</u>: Selected steps of the EOGs were compared with the corresponding portions of the FTA to determine accuracy and technical validity of conclusions.

(2) <u>PRA/human reliability analysis (HRA) Review</u>: The PRA/HRA was compared with the set of critical operator actions identified in the supporting technical material to determine whether significant human actions were selected and if the analysis was correct.

(3) <u>Summary Table Review</u>: The summary Table 12.2.1-1 was compared with the results of the FTA (e.g., available MCR I&C) for accuracy. The results were further evaluated against the I&C requirements of RG 1.97 for consistency.

ABB-CE's analysis process for the EOGs provided a large amount of specified equipment. Each step, caution, and note in the large body of EOGs was separately reviewed, analyzed, and documented through an FTA approach. A number of important controls, displays, and alarms were identified. Based on discussions between the staff and ABB-CE, it was determined that the results of the analysis would be provided in the form of an MCR ITAAC for use in the CR design implementation process. This will help ensure that the important indications, controls and alarms derived from the analysis are appropriately implemented into the HSI design. The staff has determined that ABB-CE's analysis process is acceptable.

18.9.3.3.2.2 General Results

(1) HFE Input

Although the inventory contains a list of key minimum displays, controls, and alarms necessary to carry out operator actions associated with the EOGs, ABB-CE will need to identify and further define additional detailed characteristics of these displays and controls (e.g., ranges, scales, physical dimensions, and actual information presentation) during the detailed task analysis and HSI design efforts). The staff finds that ABB-CE's HFPP, described in CESSAR-DC Section 18.4.2, including the HSI design activities and availability and suitability analyses will provide adequate assurance that these detailed characteristics are defined and implemented.

(2) Scope of the Inventory

ABB-CE developed a minimum set of fixed displays, controls, and alarms required to mitigate transients and accidents associated with the EOGs and the PRA study results. It should be noted, however, that the minimum inventory does not supersede other design requirements or commitments governing the full compliment of MCR instrumentation and controls such as federally mandated requirements (10 CFR 50.34), system I&C inventories, HFE tracking of open issues database, or additional items identified through ABB-CE's FTA of normal and abnormal operations. ABB-CE has adequately described the scope of the inventory as limited to the EOGs and the PRA study in Section 18.5.4 of the CESSAR-DC and has modified the scope of the ABB-CE Human Factors V&V Plan (Sections 6.1.4 and Sections 6.1.5) to clarify this position. The staff reviewed the revised V&V plan and found it acceptable.

18.9.3.3.3 Main Control Room Minimum Inventory Findings

The staff concluded that ABB-CE had developed an acceptable minimum set of displays, controls, and alarms that will mitigate transients and accidents associated with the EOGs and the PRA sensitivity study results. The staff determined that the discussion in Section 18.5.4 of Chapter 18, in the CESSAR-DC satisfactorily addressed the staff's concerns with regard to development of the minimum inventory. The staff's review of this issue is complete, and the minimum inventory of displays, controls, and alarms is considered adequate.

18.9.3.4 Remote Shutdown Room (RSR) Minimum Inventory

As part of the staff's review of the System 80+ design certification material, ABB-CE was requested to submit an inventory of displays, controls, and alarms necessary to permit execution of the RSR operator tasks to place and maintain the plant in a safe shutdown condition. In response to this request, ABB-CE provided an inventory for inclusion in the RSR design description and ITAAC (Table 2.12.2-1). The staff reviewed the contents of Table 12.2.2-1 and finds that it contains an adequate complement of displays, controls, and alarms necessary to place and maintain the reactor in a safe shutdown condition.

18.9.4 CDD and ITAAC Findings

The staff concludes that the System 80+ design and implementation process for HFE as described in the CDD and CESSAR-DC are acceptable. The Tier 2 commitments described in the System 80+ CESSAR-DC and related (docketed) documents provide methods and descriptions of the implementation of the Tier 1 requirements. The determination that the plant has been constructed in accordance with the design certification will require the use of the information contained in both the Tier 1 and Tier 2 documents. The Tier 2 material contained in the following System 80+ CESSAR-DC sections were used to support the safety finding with regard to the design and implementation process:

- Section 18.5, "Functional Task Analysis"
- Section 18.6, "Control Room Configuration"
- Section 18.7, "Information Presentation and Panel Layout Evaluation"
- Section 18.8, "Control and Monitoring Outside the Main Control Room"
- Section 18.9, "Verification and Validation"

Thus, as per SECY-92-287A, any change to the above CESSAR-DC section commitments by the COL applicant would involve an unreviewed safety question and, therefore, would require NRC review and approval prior to implementation. Any requested change to the subject CESSAR-DC section commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

18.10 Conclusions

The staff reviewed the HFE process described by ABB-CE in CESSAR-DC Section 18 and CESSAR-DC-referenced documents. Based on its review, the staff concludes that the ABB-CE HFE program is acceptable and provides an acceptable framework for the HSI design of the MCR, remote shutdown system, and related HSIs. The basic design features of the System 80+ advanced CR were reviewed and found consistent with human factors standards, guidelines, and principles, and acceptable for use in the CR. In addition, the staff concludes that the design commitments and the HFE ITAAC and DAC accurately summarize the minimum HFE requirements for an acceptable design and verification/validation of the MCR and remote shutdown system. All previously identified DSER issues have been adequately addressed and are resolved.

The staff finds that the HFE program described in CESSAR-DC Section 18 and CESSAR-DC-referenced documents is acceptable and will result in acceptable HSI designs for the MCR, remote shutdown system, and related applicable HSIs.

19 SEVERE ACCIDENTS

Background

Federal regulations for the design, construction, licensing, and operation of commercial nuclear power plants are defined in Chapter 1 of Title 10 of the Code of Federal Regulations (CFR). The U.S. Nuclear Regulatory Commission (NRC) evaluated the System 80+ design against these regulations, as documented in the various chapters of this report. Compliance with Federal regulations ensures that a nuclear power plant is safe enough to operate and will not impose undue risk to the general public. Compliance with these regulations also establishes the design basis of the plant.

However, the Commission also expects that new designs, like System 80+, would achieve a higher standard of severe-accident safety performance than previous designs. In an effort to provide this additional level of safety in the design of advanced nuclear power plants, the NRC has developed guidance and goals for designers to strive for accommodating events which are beyond the design basis of the plant or, in other words, beyond the requirements of traditional Federal regulations. The nuclear industry, through the Electric Power Research Institute (EPRI), has also recognized the need to establish a higher standard for advanced designs. They have developed additional standards which designers should conform to events beyond the design basis of the plant. These events are commonly referred to as "severe accidents."

When it was recognized that severe accidents needed further attention, the NRC evaluated generically, the capability of existing plants to tolerate a severe accident. It was found that the design-basis approach contained significant safety margins for the analyzed events. These margins permitted operating plants to accommodate a large spectrum of severe accidents. On the basis of this information, the Commission, in the Severe Accident Policy Statement, concluded that existing plants posed no undue risk to public health and safety and that no basis existed for immediate action on generic rulemaking or other regulatory changes for these plants because of severe-accident risk. For operating plants in the long term, the NRC developed the Integration Plan for Closure of Severe Accident Issues (SECY-88-147), in which the NRC identified the following necessary elements for closure of severe accidents:

- performance of an individual plant examination
- assessment of generic containment performance improvements (CPIs)
- improved plant operations
- a severe accident research program
- an external events program
- an accident management program

Progress continues in these areas for operating plants.

For advanced nuclear power plants including both evolutionary and passive designs, the staff concluded that vendors should address severe accidents during the design stage to take full advantage of insights gained, by designing features to reduce the likelihood that severe accidents would occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences. Incorporating insights and design features during the design phase has been demonstrated to be much more cost effective than modifying existing plants.

Regulatory Guidance

The NRC has issued guidance for addressing severe accidents. This guidance is in (1) the NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, (2) the NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants, (3) the NRC Policy Statement on Nuclear Power Plant Standardization, (4) 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants," (5) SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," and the corresponding staff requirements memorandum (SRM) dated June 26, 1990, and (6) SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," and the corresponding SRM dated July 21, 1993. Whereas, the first three documents provide guidance as to the appropriate course for addressing severe accidents, 10 CFR Part 52 contains general requirements for addressing severe accidents, and the SRMs relating to SECY-90-016 and SECY-93-087 give Commission-approved positions for implementing features for preventing severe accidents and mitigating their effects.

In SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," the staff discussed two options for proceeding with severe-accident rulemaking for the evolutionary LWR designs through the individual design certification process or generic rulemaking. In an SRM dated January 28, 1992, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe- accident issues. The effect of these actions on the System 80+ is that the criteria specified for resolving severe-accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the System 80+ design certification rulemaking as applicable regulations.

(1) Severe-Accident Policy Statement

The Commission issued the Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants on August 8, 1985. The focus of severe-accident issues in this policy statement was prompted by the NRC's judgment that accidents of this class, which are beyond the traditional design-basis events, constitute the major risk to the public associated with radioactive releases from nuclear power plant accidents. A fundamental objective of the Commission's severe-accident policy was to take all reasonable steps to reduce the chances that a severe accident involving substantial damage to the reactor core would occur and to mitigate the consequences of such an accident, should one occur. This statement described the policy that the Commission intended to use to resolve safety issues related to reactor accidents more severe than design-basis accidents (DBAs). The main focus of the statement was on the criteria and procedures the Commission intended to use to certify new designs for nuclear power plants. Regarding the decision process for certifying a new standard plant design, an approach the Commission strongly encouraged for future plants, the policy statement affirmed the Commission's belief that a new design for a nuclear power plant could be shown to be acceptable for severe-accident concerns if it met the following criteria and procedural requirements:

- demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island (TMI) requirements for new plants as reflected in the 10 CFR 50.34(f)

- demonstration of technical resolution of all applicable unresolved safety issues and the medium- and high-priority generic safety issues, including a special focus on assuring the reliability of decay heat removal (DHR) systems and the reliability of both ac and dc electrical supply systems

- completion of a probabilistic risk assessment (PRA) and consideration of the severe-accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety

- completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analyses and judgment complemented by PRA

The Commission believed that an adequate basis existed from which to establish an appropriate set of criteria. This belief was supported by the current operating reactor experience, ongoing severe-accident research, and insights from a variety of risk analyses. The Commission recognized the need to strike a balance between accident prevention and consequence mitigation and in doing so expected that vendors engaged in designing new standard plants would achieve a higher standard of severe-accident safety performance than they achieved with their previous designs.

(2) Safety Goals Policy Statement

The Commission issued the Policy Statement on Safety Goals for the Operations of Nuclear Power Plants on August 4, 1986. This policy statement focused on the risks to the public from nuclear power plant operations with the objective of establishing goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation. These are the risks from release of radioactive material from the reactor to the environment from normal operations as well as from accidents. The Commission established two qualitative safety goals that are supported by two quantitative objectives. The qualitative safety goals follow:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health, and

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative objectives were to be used in determining achievement of the above safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed, and

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plants. The Commission recognizes the importance of mitigating the consequences of a core-melt accident and continues to emphasize such features as the containment, siting in less populated areas, and emergency planning as integral parts of the defense-in-depth concept associated with its accident prevention and mitigation philosophy. The Commission approves use of the qualitative safety goals, including use of the quantitative health effects objectives in the regulatory decisionmaking process.

(3) Standardization Policy Statement

The Commission issued the Policy Statement on Nuclear Power Plant Standardization on September 15, 1987. The policy statement encouraged the use of standard plant designs and contained information concerning the certification of plant designs that are essentially complete in scope and level of detail. The intent of these actions was to improve the licensing process and to reduce the complexity and uncertainty in the regulatory process for standardized plants. In relation to severe accidents, the policy statement expected applicants for a design certification to address the four licensing criteria for new plant designs as given in the Commission's Severe-Accident Policy Statement.

(4) 10 CFR Part 52

The Commission issued 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," on April 18, 1989. This rule provides for issuance of early site permits, standard design certifications, and combined licenses with conditions for nuclear power reactors. It states the review procedures and licensing requirements for applications for these new licenses and certifications and was intended to achieve the early resolution of licensing issues and enhance the safety and reliability of nuclear power plants. Relating to severe accidents, 10 CFR Part 52 codified some of the guidance in the Severe-Accident Policy Statement and the Standardization Policy Statement. Specifically, 10 CFR 52.47 requires an application for design certification to:

- demonstrate compliance with any technically relevant portions of the TMI requirements given in 10 CFR 50.34(f)
- propose technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date 6 months prior to application and which are technically relevant to the design
- contain a design-specific probabilistic risk assessment

(5) SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements"

On January 12, 1990, the NRC staff issued SECY-90-016 in which it requested Commission approval for staff recommendations concerning proposed departures from current regulations for the evolutionary LWRs. The issues in SECY-90-016 were significant to reactor safety and fundamental to the NRC decision on the acceptability of evolutionary LWR designs. The positions in SECY-90-016 were developed as a result of (1) the NRC's reviews of current-generation reactor designs and evolutionary LWRs; (2) consideration of operating experience, including the TMI-2 accident; (3) results of PRAs of current-generation reactor designs and the evolutionary LWRs; (4) early efforts conducted in support of severe-accident rulemaking; and (5) research to address previously identified safety issues. The following preventive feature issues addressed in SECY-90-016 relating to the System 80+ were: anticipated transient without scram (ATWS), station blackout (SBO), fire protection, and interfacing-systems loss-of-coolant accident (ISLOCA). The following mitigative feature issues addressed in SECY-90-016 relating to the System 80+ were: hydrogen generation and control, core-concrete interactions (CCIs) — ability to cool core debris, high-pressure core melt ejection, containment performance, and equipment survivability. The Commission approved some of the staff positions stated in SECY-90-016 and modified others in an SRM dated June 26, 1990.

(6) SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"

On April 2, 1993, the NRC staff issued SECY-93-087 in which it sought Commission approval for staff positions pertaining to evolutionary and passive LWR design certification policy issues. This paper was an evolution of SECY-90-016. For the majority of the severe-accident issues identified in SECY-90-016, the positions in SECY-93-087 remained the same. Relative to the following two issues from SECY-90-016, the staff concluded that better definition of acceptance criteria was needed: CCIs — the ability to cool core debris (core debris coolability) and high-pressure core melt ejection. One additional containment performance issue, containment bypass potential resulting from steam generator tube ruptures (SGTRs), was identified in SECY-93-087. The Commission approved some of the staff positions from SECY-93-087 and modified others in an SRM dated July 21, 1993.

Severe-Accident Resolution

The current basis for resolution of severe accident issues for the System 80+ is 10 CFR Part 52, SECY-90-016, and SECY-93-087. The CFR (10 CFR Part 52) requires (1) compliance with the TMI requirements in 10 CFR 50.34(f), (2) resolution of unresolved safety issues and generic safety issues, and (3) completion of a design-specific probabilistic risk assessment. The staff evaluates these criteria in Sections (1) 20.3, (2) 20.1 and 20.2, and (3) 19.1 of this report, respectively.

SECY-90-016 and SECY-93-087 form the basis for the staff's deterministic evaluation of severe accident performance for the System 80+. The staff evaluates the System 80+ relative to these criteria in Section 19.2 of this chapter.

19.1 Probabilistic Safety Assessment

Executive Summary

The staff has completed the review of the System 80+ design probabilistic risk assessment (PRA) submitted by Asea Brown-Boveri-Combustion Engineering (ABB-CE) as part of its application to certify the System 80+ design under 10 CFR Part 52. ABB-CE submitted a Level 3 PRA (i.e., the PRA-calculated core damage frequencies (CDFs) as well as conditional containment failure probabilities and offsite consequences) for operation at power that addresses internal initiating events. The PRA also evaluated risk from external events (seismic, internal flood, internal fire, and tornado), as well as risk, for low-power and shutdown operation.

The staff broadened its review to encompass those safety insights that a PRA can reveal about the design, in addition to the conventional emphasis on quality and completeness of the analyses in the PRA. The staff reviewed the quality of the PRA submittal by evaluating the models, techniques, methodologies, assumptions, data, and calculational tools that ABB-CE used. In addition, it checked the PRA for completeness by comparing it with risk analyses performed for current-generation plants that had similar design characteristics. It used reported PRA results, as well as results of sensitivity, uncertainty and importance analyses, to focus the review. A sharper focus was also achieved by using PRA experience in the review process. The staff used applicable insights from previous PRA studies about key parameters and design features controlling risk. It also placed a special emphasis on PRA modeling of novel features in the design. The staff adopted this new review approach to support the multiple pre- and post-certification uses of the PRA in the 10 CFR Part 52 design certification and licensing processes. Examples are: (1) use of PRA to identify design vulnerabilities; (2) provision of PRA-based input to inspection, testing, analyses, and acceptance criteria (ITAACs); and (3) design reliability assurance program (D-RAP) and operational reliability assurance process (O-RAP). In this regard, this new approach is consistent with the objectives and intent of 10 CFR Part 52.

The PRA findings and insights about the System 80+ design which resulted from the staff's review are reported in this chapter. Specifically, the following are discussed: (1) the special evolutionary "preventive" and "mitigative" features that have been incorporated into the System 80+ design (Section 19.1.2), (2) major safety insights from the internal events analysis for operation at power (Section 19.1.3), (3) major safety insights from the external events analysis for operation at power (Section 19.1.4) include insights from the PRA-based seismic margins analysis (SMA) (Section 19.1.4.1), the internal fires analysis (Section 19.1.4.2), the analysis for internal flooding (Section 19.1.4.3), and the tornado strike analysis (Section 19.1.4.4), (4) major safety insights for operation at low power and during plant shutdown (Section 19.1.5), (5) the staff's evaluation on the use of PRA in the design process (Section 19.1.6), and (6) PRA input to the design certification process (Section 19.1.7). These PRA-based findings and insights are summarized below.

Special Evolutionary "Preventive" and Mitigative" Design Features

The several special features and their functions are qualitatively discussed first. These design features were introduced into the System 80+ design for the purpose of reducing the CDF and the conditional containment failure probability (CCFP) from internal events (as compared to System 80 design and currently operating nuclear power plants). It should be noted that this introductory discussion is not based solely on PRA findings or insights.

The larger pressurizer and steam generators (SGs) in the System 80+ design make the plant's response to transients slower and less severe (e.g., lower temperature and pressure peaks are reached), thus contributing to the prevention and mitigation of transients by reducing the number of plant transients and arresting their progression once started. The incorporation of shutdown cooling system (SCS) pumps that are functionally interchangeable with the containment spray system (CSS) pumps contributes to the increased availability of these two important front-line systems to perform their intended functions. The multiple independent connections to the offsite electrical grid combined with the turbine-generator runback capability to maintain hotel loads contribute to the reduced frequency of loss of offsite power (LOOP) initiating events. The provision of two EDGs (each with dedicated batteries) and a standby combustion turbine generator (CTG) combined with the provision for six vital batteries contribute to the high reliability of the emergency ac and dc power sources that reduce the frequency of SBO sequences. The use of separate startup and emergency feedwater systems (EFWSs) help reduce the demands on the EFWS. In addition, the four-train dedicated safety EFWS (two motor-driven and two turbine-driven pumps) provides redundancy which reduces the failure probability of secondary-side heat removal. The use of two turbine-driven pumps for supplying emergency feedwater (EFW) to the SGs helps reduce the CDF of SBO sequences. The four-train safety injection system (SIS) increases the reliability of this system to levels above those for current-generation pressurized-water reactor (PWR) plants by: (1) decreasing the system's unavailability due to outages (testing, repair, maintenance, etc.); (2) eliminating the low-pressure pumps, thus eliminating the failures to start for these pumps; and (3) eliminating the need to realign the suction of the pumps.

The incorporation of the rapid depressurization (RD) capability into the safety depressurization system (SDS) provides a manual safety-grade means of rapidly depressurizing the reactor coolant system (RCS) so that the SIS can be actuated for feed-and-bleed operation, when the long-term DHR fails via either the SCS or the SG secondary-side heat removal. The System 80+ PRA shows that this is a very important feature which helps reduce the failure probability of long-term DHR. In addition, the RD function of the SDS also serves a mitigative function. Specifically, the actuation of the SDS, before the core debris penetrates the vessel, can reduce or eliminate the potential for direct containment heating (DCH) and large hydrogen combustion events at vessel breach and thus can reduce the probability of early containment failure.

The in-containment refueling water storage tank (IRWST) eliminates the need for switching over from the injection mode to the recirculation mode during emergency core cooling operations. The IRWST is also important to the progression of a severe accident within the containment because of its ability to reduce containment pressure (through steam condensation when releases are into the IRWST), to reduce fission-product release (through pool scrubbing), and reduce the probability of CCI through reactor cavity flooding. The large spherical steel containment, in addition to the high-pressure capacity, improves containment atmospheric mixing and dilution of postaccident hydrogen gases, thereby reducing the potential for developing detonable concentrations of hydrogen under severe-accident conditions.

The System 80+ reactor cavity design is such that only a small fraction of the core debris discharged from the reactor vessel in a high-pressure vessel breach would be dispersed to the upper compartment of the containment, thereby reducing the potential for containment failure from DCH. The reactor cavity is also designed to enhance ex-vessel debris coolability, by providing a large floor area for debris spreading and a dedicated system for flooding the reactor cavity and debris with water. Finally, the design incorporates an ignition system to promote combustion at lean hydrogen concentrations and minimize the potential for large deflagrations or detonations.

<u>Major Safety Insights From the Internal Events PRA for Operation at Power</u>

ABB-CE estimated the CDF from internally initiated events at approximately 2×10^{-6}/year. This is approximately 50 times smaller than the CDF estimate for the System 80 design from which the System 80+ design has evolved. For System 80, LOOP including SBO (LOOP/SBO) essentially dominates the CDF profile from internal events (~46 percent of total CDF). This is not the case for System 80+, for which the contribution from LOOP/SBO is ~2 percent. This decrease is due to the following System 80+ design features: (1) two physically separate and electrically independent switchyards, (2) turbine-generator runback (steam bypass) capability to maintain hotel loads on a loss of grid, (3) addition of the standby CTG (onsite alternate ac (AAC) power source), (4) the six Class 1E 125-V batteries with over 8-hour SBO coping capability, and (5) four-train EFWS, including two turbine-driven pumps. For the System 80+, the LOCA categories of initiating events (~37 percent contribution) and the "transient" events category (~35 percent contribution) are leading contributors to the total CDF from internal events. They are followed by the SGTR event (~17 percent) contribution. The largest reductions in CDF with respect to the System 80 design are associated with sequences initiated by LOOP, "transients," small LOCAs, and SGTR events (the specific evolutionary features that contributed to these reductions are listed in Section 19.1.3.1.2 of this chapter).

An uncertainty analysis was performed to determine the magnitude of uncertainties that characterize the CDF estimates as well as the major contributors to these uncertainties. Only uncertainties associated with reliability and availability data were considered. Modeling uncertainties, which are generally not considered in PRA studies, were not accounted for. Insights from the uncertainty analysis indicate that no large data-related uncertainty is associated with the total (internal events) CDF estimate. Most of the major contributors to the dominant accident sequences (and total CDF) have relatively small data-related uncertainties associated with them. Exceptions are: (1) operator failure to perform an "aggressive secondary cooldown" to depressurize in order to use the low-pressure SCS for injection when the SIS fails during a LOCA or SGTR event; (2) operator failure to initiate feed-and-bleed operation to provide an alternate DHR path when heat transfer through the SGs or the SCS is unavailable; and (3) common-cause failure (CCF) of all SIS pumps to start.

An importance analysis was performed that addressed two general objectives: (1) insights on risk reduction and (2) insights on safety or reliability assurance. The first objective was achieved by the identification and ranking of systems, structures, and components (SSCs), as well as human actions and initiating events that are major contributors to the estimated CDF from internal events (i.e., having high "risk reduction worth"). This ranking was very useful in the design and design certification processes. ABB-CE identified the areas in which plant risk could be reduced by design changes and operational requirements, such as improved testing and maintenance for SSCs and improved training and procedures for human actions. The second objective was achieved by the identification and ranking of SSCs as well as human actions which are the major contributors to maintaining the "built-in" safety level of the System 80+ design (i.e., having high "risk achievement worth"). This ranking was very useful in identifying areas in which it is particularly important to implement the design and operational requirements assumed during the System 80+ design development and design certification processes (such as ITAAC, D-RAP, O-RAP, technical specifications (TS), operator training and procedures) to avoid unacceptable risk increases.

Insights from the importance analysis show that events which would decrease significantly the built-in safety level of the System 80+ design (i.e., those associated with SSCs or human actions having high "risk achievement worth") are hardware CCFs and human errors. This is due to the redundancy and diversity of the System 80+ safety systems, which ensure that single component hardware failures are not among those events whose occurrence would have a large impact on the CDF from internal events. CCFs of sets of components with large impact on the estimated CDF, i.e., sets of components with high risk achievement worth are (1) electrical distribution system (EDS) components, such as 125-V dc Class 1E buses, the 480-V ac load transformers, and the 4.16-kV Class 1E buses, (2) EFWS components, such as the distribution line check valves and the pump discharge check valves, (3) SIS components, such as the safety injection (SI) line check valves and isolation valves, and the SIS pumps, (4) SDS components, such as the RD ("bleed") valves and the Class 1E dc power source (through dedicated dc-ac inverters), and (5) SCS components, such as the discharge check valves and the motor-operated isolation valves. Operator errors found to have a large impact on the estimated CDF (i.e., errors associated with operator actions having high risk achievement worth) are: (1) operator failure to initiate hot-leg injection to prevent boron crystallization during a medium or a large LOCA; (2) operator failure to align the condensate storage tank (CST) to the emergency feedwater storage tanks (EFWSTs) to provide makeup water and to continue DHR from the reactor core; and (3) operator failure to initiate primary feed-and-bleed operation when secondary heat removal is unavailable. Single-component failures which have a significant impact on the estimated CDF, i.e., single components with significant risk achievement worth are those of a manual isolation valve and a check valve in the line between the CST and the EFWSTs.

Insights from the importance analysis show that failures of components associated with the following events are major contributors to the estimated CDF from internal events (i.e., they have the highest risk reduction worth): (1) initiating events, such as loss of main feedwater, medium and small breaks (LOCAs), and SGTRs, (2) CCF of the SIS injection line check valves, (3) CCF of the EFWS pump check valves or the EFWS distribution line check valves, and (4) CCF of the RD ("bleed") valves. In addition, operator actions with highest "risk reduction worth" are (1) performance of an "aggressive secondary cooldown" when SIS is unavailable during a small LOCA or an SGTR accident and (2) initiation of primary feed-and-bleed operation when secondary heat removal is unavailable.

ABB-CE performed sensitivity analyses to determine the sensitivity of the estimated CDF to potential biases in numerical values, potential lack of modeling details, and to previously raised issues. An insight drawn from the sensitivity analyses is that although the estimated CDF is less sensitive to human error probabilities than operating reactor designs, it is still very sensitive to operator actions which are carried out outside the main control room (MCR) during an accident. Another insight is that the estimated CDF is very sensitive to several CCF probabilities. This underlines the importance of those design features and operational requirements that aim at preventing CCFs, such as divisional separation, diversity of some redundant components, and appropriate maintenance programs. The CDF estimate for the System 80+ design is not very sensitive to reasonable changes in single-component failure probabilities or initiating event frequencies. This is a consequence of the redundancy and diversity built into the System 80+ design. Sensitivity analysis indicates that the CDF estimate is not sensitive to the probability of a reactor coolant pump (RCP) seal failure after an SBO or loss-of-cooling-water event. This results from the reduced likelihood of SBO events and from the increased reliability of RCP seal cooling for the System 80+ design as compared to operating reactor designs. Finally, sensitivity analysis indicates that the System 80+ CDF estimate is not very sensitive to increases in failure rates of motor-operated valves (MOVs).

The results of the Level 2 and 3 portions of the System 80+ PRA show that the System 80+ containment is quite robust and able to accommodate severe-accident challenges with a low attendant probability of containment failure. In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity and (2) releases which result in doses of 0.25 Sv (25 rem) (whole body) or greater at 0.8 km (0.5 miles) from the reactor. Using the structural integrity definition of containment failure, the conditional CCFP for the System 80+ is approximately 11.7 percent for internal events and 11.4 percent when tornado strike events are added to the internal events. In ABB-CE's analysis, many of the containment failures are associated with containment basemat melt-through and occur well after 24 hours. If such sequences are ignored, the CCFP would be approximately 4.3 percent (for internal plus tornado strike events). If doses in excess of 0.25 Sv (25 rem) are considered to constitute containment failure (i.e., the dose definition of containment failure), then the CCFP would be 2.7 percent.

The staff concludes that the estimated CCFP for the System 80+ design conforms to the Commission's containment performance goal (CPG) (i.e., 10 percent). Specifically, within the 24-hour period after core damage, which is the focus of the CPG, the probability of containment failure (using either the structural integrity or the dose definition of containment failure) is below the goal. However, the CCFP is somewhat higher than the goal (11 percent) when failures beyond 24 hours are included and the structural integrity definition of failure is used. The CCFP remains less than the goal when the EPRI-based dose definition of failure is used (2.7 percent). In SECY-90-016, the staff stated that in view of the low probability of accidents that would challenge the integrity of the containment, the CCFP for evolutionary designs should not exceed "approximately" 0.1 (10 percent). Furthermore, in the related SRM, the Commission directed the staff that the CCFP objective of 0.1 should not be imposed as a requirement in and of itself. In view of the approximate nature of the CPG, the recognition that PRA results, particularly bottom-line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11-percent CCFP estimate are late containment basemat melt-throughs rather than releases to the atmosphere, the staff concludes that the System 80+ design satisfies the Commission's CPG.

On the basis of the Level 3 PRA, the estimated total risk to the public posed by the for System 80+ design is quite small. ABB-CE's analysis indicates a total dose of about 17 person-rem over a 60-year plant life. Total risk is dominated by events which lead to bypass of the containment (primarily SGTR events), and early containment failure. This is consistent with results from PRAs for operating plants.

Major Safety Insights from the External Events PRA for Operation at Power

System 80+ is designed to withstand a 0.3 g safe-shutdown earthquake (SSE). Since the analyses used in designing the capability of SSCs to withstand the SSE contain significant margin, it is expected that a plant built to withstand the SSE actually will be able to withstand a much larger earthquake. A PRA-based margins analysis systematically evaluates the capability of the designed plant to withstand earthquakes without sustaining core damage, but does not estimate the CDF from seismic events. The margins analysis is simply a somewhat conservative method for estimating the "margin" above the SSE, that is, how much larger than the SSE an earthquake must be before it compromises the safety of the plant.

The capability of a particular SSC to withstand beyond-design-basis earthquakes is measured by the value of the peak ground acceleration (g-level) at which there is a high confidence that the particular SSC will have a low probability of failure (HCLPF). The HCLPF capacity of a certain SSC corresponds to the earthquake level at which, with high confidence (95 percent), it is unlikely (probability less than 5×10^{-2}) that the SSC will fail. An HCLPF value for the entire plant is determined by finding the lowest sequence HCLPF that leads to core damage. It is a measure of the capability of the plant to withstand beyond-design-basis earthquakes without sustaining core damage. The plant HCLPF value, which is assessed from the SSC HCLPF values, has units of acceleration. The NRC has indicated that it expects that a plant truly designed to withstand a 0.3 g SSE should have a plant HCLPF at least 1.67 times the SSE (i.e., 0.5 g). The margins analysis has shown that the System 80+ design meets (and exceeds) the 0.5 g HCLPF value expectation.

The PRA-based SMA, performed by ABB-CE and reviewed by the staff, identified 31 sequences that lead to core damage. In all of these sequences, offsite power is lost. The most important sequence involves the seismic gross structural failure of the containment (HCLPF value of 0.73 g). If random failures and human errors are not taken into account (i.e., when cutsets containing seismic failures only are considered), the plant HCLPF value was estimated by ABB-CE to be 0.73 g. Since the plant HCLPF value can be lower when certain random failures (or human errors) occur simultaneously with the seismic failure of certain SSCs, cutsets containing both seismic and non-seismic failures were examined to find out if there were any cutsets with a combination of HCLPF below 0.5 g and random (or human errors) greater than 1×10^{-3}. The most significant cutset was identified to be the product of two events: (1) seismic failure of the standby CTG (which has an HCLPF value of 0.36 g) and (2) random failure of both EDGs (which has a probability of about 1×10^{-2}). ABB-CE performed sensitivity analyses to evaluate the effects of changes in certain assumptions used in the analysis, for example, HCLPF values for key SSCs and changing of the site conditions from rock to various soil types. These analyses are helpful in determining which SSCs should be added to the D-RAP and O-RAP. The margins analysis pointed out the need for the standby CTG to be procured and installed with a robust capability to withstand seismic events. The same analysis showed the importance of minimizing random failures and unavailabilities of both EDGs.

ABB-CE performed a "scoping" quantitative risk analysis for internal fires in conjunction with a qualitative fire analysis to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. This scoping analysis considered fires in the nuclear annex and the station service water/component cooling water (SSW/CCW) building. Fires in the MCR or in the containment were examined separately using both qualitative and quantitative arguments. The staff finds that the System 80+ design has significant robustness to prevent and mitigate severe accidents initiated by fires and should result in a plant with superior capabilities to prevent and mitigate fires compared to operating nuclear power plants. The most important design features contributing to the reduced likelihood of a fire leading to core damage in the System 80+ design, as compared to operating plants, are: (1) a reinforced-concrete wall between the two safety divisions that serves as a floor-to-ceiling barrier rated for at least a 3-hour fire with no doors up to elevation 70 ft; (2) the capability of using the remote shutdown panel to respond to a transient or accident (in the unlike event of a fire in the MCR requiring its evacuation); (3) elimination of the cable spreading room; and (4) 3-hour-rated fire barriers that are seismic Category I and are made of reinforced concrete with doors that automatically close and are alarmed in the MCR.

ABB-CE performed a scoping quantitative risk analysis for internal floods in conjunction with a qualitative flood analysis to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. The most important features which contribute to the reduced likelihood of a flood leading to core damage in the System 80+ design, as compared to operating plants, are (1) the use of only closed cooling water systems in the nuclear annex (there is no cross-connection between the divisions of the component cooling water system (CCWS)), (2) the reinforced-concrete wall between the divisions in the nuclear annex that has no doors or passages below elevation 70 ft, (3) flood barriers between the quadrants in the subsphere to help limit internal floods to one quadrant, (4) only limited sources of water within the nuclear annex and no paths through which water from external "unlimited" sources can enter the nuclear annex or reactor building (RB), and (5) reinforced concrete walls in the SSW/CCW structure separating divisions to protect against interdivisional floods.

ABB-CE assessed the System 80+ CDF from tornados and estimated it to be about 3×10^{-7} per year. It was conservatively assumed in the analysis that a tornado strike event at the site would result in (1) a LOOP with a duration greater than 24 hours, (2) the loss of the turbine-generator runback capability to pick up hotel loads, and (3) the loss of the non-safety-grade CTG. The implication of these assumptions is that, after the tornado strike, the EDGs are required to operate for at least 24 hours. Important design features contributing to the lower CDF from tornados for System 80+, as compared to operating nuclear power plants, are (1) no safety-related equipment in the turbine building, (2) use of reinforced concrete for the station service water (SSW) intake structure, and (3) two EFWSTs, the primary sources of EFW for DHR, which are located in the nuclear annex (a reinforced-concrete structure) and below grade level. An important safety insight gained from the tornado analysis is that "blockage of the service water intake flow by tornado-generated debris" is a major contributor to risk from tornados. The COL applicant should evaluate the vulnerability of the SSC intake structure to tornado-generated debris. This is COL Action Item 19-1.

<u>Major Safety Insights From the Risk Analysis for Low Power and Shutdown (LP&S) Operation</u>

ABB-CE assessed the risk associated with LP&S operation for Mode 4 (hot shutdown), Mode 5 (cold shutdown), and Mode 6 (refueling) for internal fires, internal floods, and other internal events. The major objectives were: (1) identify design and operational vulnerabilities related to LP&S operation and (2) identify risk-important design features, plant configurations, human actions, and operational requirements. The System 80+ design CDF from internally initiated events, during shutdown operation, was estimated to be approximately 6 × 10^{-7}/year. With respect to initiating events, LOOP is the leading contributor to the estimated CDF (~39 percent), followed by loss of DHR (~36 percent) and LOCA (~25 percent). With respect to plant configurations, the leading contributor is Mode 5 with reduced inventory (~48 percent contribution), followed by Mode 6 with the IRWST empty (~30 percent contribution). The third leading contributor (~20 percent contribution) is the plant configuration which includes Modes 4 and 5 with normal inventory and Mode 6 with the IRWST full (primarily due to the increased likelihood of LOOP because of the long interval spent in this configuration).

The following are important factors contributing to the decrease or elimination of vulnerabilities in the System 80+ design during shutdown operation, as compared to currently operating nuclear power plants: (1) defense-in-depth approach that provides alternative means for maintaining coolant inventory and removing decay heat during a LOCA or a loss-of-shutdown-cooling (loss-of-DHR) event, (2) design features and operational requirements for preventing and mitigating LOOP/SBO events, (3) COL requirements for minimizing risk associated with human errors through appropriate outage management, administrative controls, procedures, training, and knowledge of plant configuration, and (4) COL requirements for configuration control to ensure the integrity of fire and flood barriers between areas in the same division (e.g., quadrants) where systems comprising the alternate shutdown success paths are located.

Important design features and operational requirements for defense in depth during shutdown are (1) two separate and independent divisions of the SCS whose pumps are identical to and functionally interchangeable with the CSS pumps (this characteristic contributes to the increased availability of these two systems, as compared to operating reactor designs, to maintain coolant inventory and/or remove decay heat), (2) the capability to align the SCS to the IRWST during plant shutdown operation to provide inventory makeup or to perform a feed-and-bleed operation, (3) if all SCS/CSS pumps are unavailable for DHR and/or coolant inventory makeup, the operator can still perform these functions by feed-and-bleed operation using the SDS or the low-temperature overpressure protection (LTOP) valves for the bleed function and the SIS or the chemical and volume control system (CVCS) pumps for the feed function, and (4) a new TS, added as a result of PRA insights, requiring that two of the four SIS pumps be available in shutdown modes when the IRWST is available (because of the importance of SI for feed-and-bleed operation during shutdown).

Design features which are important for preventing and mitigating LOOP/SBO events during power operation, are also important in reducing the frequency of these events during shutdown operation. These features are: (1) two separate and independent switchyards and (2) redundant and diverse onsite ac power sources (two Class 1E EDGs and a non-safety-grade CTG). Important operational requirements during shutdown operation are (1) a new System 80+ TS, added as a result of PRA insights, requiring that two of the three onsite power sources (i.e., two EDGs and one CTG) be available during shutdown operation (reduced RCS inventory) and (2) assurance by the COL applicant that, when a switchyard is unavailable for maintenance, no activities which could fail the operating switchyard are taking place and no fire sources are present. This is COL Action Item 19-17.

ABB-CE performed an importance analysis for LP&S operation; the objectives of this analysis were similar to the objectives for the power operation analysis, i.e., risk reduction and safety or reliability assurance. Because the models for system failures and interactions during shutdown are less detailed than the models for power operation, the importance analysis was performed at the system or function level rather than at the component or event level. Nevertheless, important insights were drawn from this analysis. The identification of system functions, operator actions, and initiating events which have high risk reduction worth, provided important insights on areas in which the plant risk could be reduced by design changes and operational requirements, such as TS, improved testing and maintenance, improved and procedures. The identification of system functions and operator actions having high risk achievement worth, helped identify certification and operational requirements (such as ITAACs, maintenance, training, outage management, configuration control, and procedures) which ensure that the "built-in" design reliability will be maintained during construction and operation. ABB-CE also performed sensitivity analyses for LP&S operation. Important insights from these analyses are (1) the CDF estimate is sensitive to changes in the frequency of accident-initiating events during shutdown, such as LOOP, LOCAs, loss of DHR, and internal fires (these insights demonstrate the importance of appropriate outage management programs to minimize mistakes which cause these initiating events to happen), and (2) reduced RCS inventory is the most critical operation during shutdown (use of SG nozzle dams for SG maintenance and inspection, as a method of limiting the time spent in this configuration, has a positive effect on the estimated CDF for shutdown operation).

Use of PRA in the Design Process

ABB-CE used PRA in the design process to achieve the following objectives: (1) identify and quantify vulnerabilities in operating reactor designs and introduce features and requirements that reduce or eliminate these vulnerabilities, (2) quantify the effect of new design features and requirements on plant risk in order to confirm the risk reduction credit for these improvements, and (3) select among alternative design features or design options. ABB-CE used PRA insights from both operating reactor experience and the System 80 design (from which the System 80+ design evolved), to identify potential vulnerabilities in operating reactor designs. This information was used to introduce "evolutionary" design features and make the transition from the System 80 to the System 80+ design. Once these features were introduced, PRA was used to quantify their effect on risk and confirm acceptable reduction or elimination of vulnerabilities, including compliance with applicable risk goals.

PRA Input to the Design Certification Process

PRA was used in the design certification process to achieve the following objectives: (1) develop an in-depth understanding of design robustness and tolerance of severe accidents initiated by either internal or external events, (2) develop a good appreciation of the risk significance of human errors associated with the design, and characterize the key errors in preparation for better training and refined procedures, and (3) identify important safety insights and assumptions to support certification requirements, such as ITAACs, D-RAP and O-RAP requirements, TS, as well as COL and interface requirements. The first two objectives were achieved by identifying the dominant accident sequences as well as the risk-important design features (SSCs) and human actions (see Sections 19.1.3 to 19.1.5 of this chapter). The third objective was achieved by using PRA insights and assumptions to develop a list of design certification requirements (see Section 19.1.7 of this chapter).

Conclusions and Findings

The NRC staff has evaluated the quality of the PRA performed by ABB-CE for the System 80+ design as well as the use of PRA in the design development and in the design certification process. The staff concludes that the quality and completeness of the System 80+ PRA are adequate for its intended purposes, such as supporting the design and the design certification process. The approaches used by ABB-CE for both the core-damage and containment analyses are reasonable and sufficient to achieve the desired goals of describing and quantifying potential core-damage scenarios and containment performance during severe accidents. The staff concludes that the use of PRA in the design process helped introduce improved or unique evolutionary features (such as the RD capability of the SDS, the IRWST, and the reactor cavity flooding system (CFS)) that contributed to the reduced CDF and CCFP estimates of the System 80+ design when compared with those of operating PWRs. PRA results and insights were used to identify areas in which it is particularly important to implement the design and operational requirements assumed for the design certification (e.g., ITAACs, D-RAP, O-RAP, TS, operator training, and procedures). On the basis of this review, the staff finds that the System 80+ design represents an improvement in safety over operating PWRs in the United States.

19.1.1 Introduction

As part of the System 80+ evolutionary design certification application, ABB-CE submitted a PRA in accordance with the requirements of 10 CFR 52.47 and the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (as described in FR Vol. 50, No. 153, dated August 8, 1988, p. 32138 dated August 8, 1991). The NRC staff's assessment consisted of the traditional evaluation of events that could lead to core damage and offsite consequences as well as an evaluation of what the PRA revealed about the System 80+ design.

19.1.1.1 Background and NRC Review Objectives

The general objectives of the NRC staff's review of the System 80+ PRA were to (1) identify safety insights based on systematic risk-based evaluations of the design; (2) determine in a quantitative manner whether the design represents a reduction in risk over existing plants; (3) examine the balance of preventive and mitigative features of the design; (4) assess the reasonableness of the risk estimates documented in the PRA; and (5) support design certification requirements, such as inspections, tests, analyses, and acceptance criteria (ITAACs), design reliability assurance program (D-RAP) and operational reliability assurance process (O-RAP), technical specifications (TS), as well as COL and interface requirements. In addition, the staff used the System 80+ PRA to determine how the risk associated with the design relates to various safety goals and to discover design and procedural vulnerabilities.

The objectives are drawn from 10 CFR Part 52, the Commission's Severe Reactor Accident Policy Statement regarding future designs and existing plants, the Commission's Safety Goal Policy Statement, the Commission approved positions concerning the analyses of external events contained in SECY-93-087, and NRC interest in the use of PRA to help improve future reactor designs. In general, these objectives have been achieved by the System 80+ PRA and the NRC staff's review. The staff's proposed applicable regulation for the analysis of external events for the System 80+ PRA is as follows:

> The application for design certification must contain a probabilistic risk assessment that includes an assessment of internal and external events. Simplified probabilistic methods and margins methods may be used to assess the capacity of the standard design to withstand the effects of external events such as fires and earthquakes. Seismic margin analysis must consider the effects of earthquakes with accelerations approximately one and two-thirds the acceleration of the SSE.

During the construction stage, the COL holder will be able to consider as-built information. The Commission believes that updated PRA insights, if properly evaluated and utilized, could strengthen programs and activities in areas such as training, development of emergency operating procedures (EOPs), reliability assurance, maintenance, and 10 CFR 50.59 evaluations. The plant-specific PRA developed from the design certification should be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, TS, plant-specific EOPs and design changes (Ref. 1). This is COL Action Item 19-12. These updates are the responsibility of the COL applicant or COL holder. As plant experience data accumulate, failure rates (taken from generic data bases) and human errors assumed in the design PRA are to be revised and incorporated, as appropriate, into the O-RAP ((SECY paper memorandum, Samuel J. Chilk (NRC) to James M. Taylor (NRC), "SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," July 21, 1993). This is COL Action Item 17-4.

19.1.1.2 Evaluation of PRA Quality and Resolution of Open Issues

The NRC staff has completed its review of the quality and completeness of the System 80+ PRA. These attributes are essential in using the PRA to draw insights about the design robustness and tolerance to severe accidents and to provide risk-based input to pre- and post-certification activities, thus achieving the objectives itemized above (Section 19.1.1.1). The staff reviewed the quality of the PRA submittal by evaluating the models, techniques, methodologies, assumptions, data, and calculational tools that were used by ABB-CE. In addition, the staff checked the PRA for completeness by comparing it with risk analyses performed for current-generation plants with similar design characteristics.

The review of the quality and completeness of the PRA submittal involved the issuance of requests for additional information (RAIs) to ABB-CE, followed by the evaluation of ABB-CE's responses to the RAIs. In conducting the technical review, the staff followed guidance similar to that in the "PRA Review Manual" (NUREG/CR-3485). It used reported PRA results as well as results of sensitivity, uncertainty, and importance analyses to focus the review. A sharper focus was also achieved by using PRA experience in the review process. The staff used applicable insights from previous PRA studies about key characteristics and design features controlling risk. The staff also placed a special emphasis on PRA modeling of novel features in the design.

Although the review has been a continuous process, it involved two distinct stages. The first stage of the review ended with the issuance of a draft safety evaluation report (DSER). In the DSER, three classes of items were identified that the staff believed needed additional attention by ABB-CE. The classes were (1) open items, that is, areas where the staff disagreed with the submittal or required additional supporting documentation; (2) confirmatory items, that is, areas in where the staff and ABB-CE agreed on a proposed resolution but additional documentation was required; and (3) COL action items, that is, areas where the COL applicant should factor in plant- or site-specific information at the COL stage. The second stage of the review involved the resolution of all DSER open and confirmatory items, the inclusion of all identified COL action items, and the preparation of the final safety evaluation report (FSER). The resolution (closure) of DSER open items involved close interaction between the staff and ABB-CE, including several rounds of RAIs and ABB-CE's responses. A summary of DSER issues and the associated resolutions is given in Appendix 19A of this chapter.

The NRC staff concludes that the quality and completeness of the System 80+ PRA are adequate for its intended purposes, such as supporting the design and certification processes. The approaches used by ABB-CE for both the core-damage and containment analyses are logical and sufficient to achieve the desired goals of describing and quantifying potential core-damage scenarios and containment performance during severe accidents. All open items reported in the DSER as well as all followup issues were resolved satisfactorily.

ABB-CE submitted an update to Chapter 19 that revises CESSAR-DC Table 19.15-1 and contains additional PRA insights that were agreed upon in a public meeting held on

January 31, 1994. In addition, ABB-CE added a new CESSAR-DC Section 19.15.6. This new section contains a table of risk-important structures, systems, and components (SSCs) for the D-RAP, a table of PRA-important operator actions, and several COL action items. The staff has reviewed these changes and finds them acceptable. These changes were incorporated in the CESSAR-DC in Amendment V. Therefore, this is part of FSER Confirmatory Item 1.1-1 (see Chapter 1 of this report) is resolved.

The special evolutionary features that were incorporated into the System 80+ design for the purpose of preventing and mitigating accidents are briefly presented in Section 19.1.2 below. Safety insights about the System 80+ design, drawn from the internal events risk analysis for operation at power, are presented in Section 19.1.3. Safety insights from the external events risk analysis (seismic, internal fires, internal floods, and tornado strikes) for power operation are reported in Section 19.1.4. Safety insights associated with low-power and shutdown operation are reported in Section 19.1.5. Section 19.1.6 reports the use of PRA in the design process, while Section 19.1.7 presents the PRA input (derived from PRA insights and assumptions) to the design certification process. Finally, Section 19.1.8 summarizes the major conclusions and findings about the design consistent with the objectives of the PRA and its use in the design and certification processes.

19.1.2 Special Evolutionary Design Features

The System 80+ standard design evolved from the System 80 design through incorporation of several design changes intended to make the plant safer, more available, and easier to operate. Insights from the System 80 PRA, as well as from previous PRAs for operating reactors, helped identify these design changes. Therefore, the System 80+ design contains features that improve plant safety, and thus reduce risk, when compared to the current-generation nuclear power plants.

Some of these special evolutionary design features are preventive in nature, others are mitigative. Preventive features aim to (1) minimize the initiation of plant transients, (2) arrest the progression of plant transients once they start, and (3) prevent severe accidents (core damage). Mitigative features aim to mitigate severe accidents, that is, the consequences of core damage. The major preventive and mitigative evolutionary design features of the System 80+ design are described in Sections 19.1.2.1 and 19.1.2.2, respectively. In these descriptions, a brief qualitative discussion points out the effect that each of these evolutionary features has on various elements involved in severe-accident prevention and mitigation. More details about these features are found in the appropriate chapters of the CESSAR-DC.

19.1.2.1 Evolutionary Design Features for Preventing Core Damage

The following major features were incorporated into the System 80+ design for the purpose of limiting plant transients and preventing severe accidents:

Larger Pressurizer

The reason for the larger pressurizer volume of the System 80+ design, as compared with the existing generation of commercial nuclear power plants, is to make the plant response to transients slower and less severe, thus allowing more time for operator actions. The larger volume helps maintain higher pressurizer pressure and water level after a turbine trip, thus increasing the margin to pressurizer safety valve challenges.

It also helps prevent the emptying of the pressurizer after overcooling transients, thus increasing the margin for a SI actuation signal. For certain transient events, such as loss of main feedwater, the rise in pressurizer pressure is moderated, thus reducing the likelihood of challenging the primary safety valves (PSVs). A larger pressurizer volume also helps lower the peak pressure that can be reached after a postulated ATWS event.

Larger Steam Generators

The increased heat transfer area of the System 80+ design SGs, as compared with operating reactors, provides a 10-percent tube plugging margin. The increased secondary-side volume (and inventory) makes transients slower and also increases the boil-off time to dry out the SGs, thus extending the time available to the operators for recovery actions. The corrosion-resistant SG tube materials and the reduced hot-leg temperature are expected to help reduce the frequency of SG tube ruptures.

Functionally Interchangeable SCS and CSS Pumps

The SCS and the CSS are integrated, and the SCS and CSS pumps are designed to be independent but identical and functionally interchangeable, thus ensuring backup and higher reliability for both systems. In addition to their long-term DHR function, the SCS pumps can be used to back up the SIS to inject borated water into the reactor core (in conjunction with the rapid depressurization system (RDS). The SCS pumps can also be used to back up the CSS pumps for cooling the IRWST during feed-and-bleed operations.

SIS With Four Trains and Direct Vessel Injection (DVI)

The SIS is a dedicated four-train safety system whose primary function is to inject borated water into the RCS for inventory and/or reactivity control during severe accidents, such as LOCAs and ATWS. It can also be used in conjunction with the SDS for feed-and-bleed operation. For continuous long-term postaccident cooling of the core, the SIS pumps are manually realigned for simultaneous hot-leg and DVI to prevent boron crystallization. The major evolutionary characteristics of the System 80+ SIS are (1) four high-pressure 100-percent capacity pumps (a) four-train, as compared to a two-train, SIS improves system reliability and reduces the contribution to the system unavailability that is due to outages for testing and maintenance) and (2) elimination of the need for low-pressure pumps (eliminates failures associated with starting these pumps in operating reactor designs).

Safety Depressurization System

An important function of the SDS is to serve as the manual safety-grade means of rapidly depressurizing the RCS so that SI can be actuated, when DHR fails via either the SGs or the SCS, to cool the core by feed-and-bleed operation. The RD function of the SDS constitutes the "bleed" portion of the feed-and-bleed operation, and SI constitutes the "feed" portion. This is an important feature added to the System 80+ design to reduce the failure probability of long-term DHR. The SDS also has a mitigative function. RD can be used to mitigate some of the potential containment challenges associated with reactor vessel failure at high pressure (see Section 19.1.2.2 below).

Multiple Independent Connections to the Grid and Turbine-Generator Runback Capability

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine-generator system and the associated buses are designed to run back to maintain hotel loads on a loss of grid. The purpose of these features is to reduce the frequency of loss-of-offsite-power (LOOP) initiating events and therefore the frequency of accident sequences that are associated with LOOP including SBO.

Separate Startup and Emergency Feedwater System (EFWS) and Four-Train EFWS

The use of a non-safety-grade startup feedwater system (SFWS) for normal startup and shutdown operations helps reduce the demands on the EFWS. In addition, the SFWS serves as an independent means of supplying feedwater to the SGs for removing heat from the RCS during emergency conditions when main feedwater is not available. The EFWS is a dedicated system, that serves as an independent safety-related means of supplying feedwater to the SGs for the early phase of DHR if normal feedwater is lost. The EFWS consists of two trains, each aligned to feed its respective SG. Each train contains one motor-driven pump subtrain and one turbine-driven pump subtrain. For SBO sequences the turbine-driven EFW pumps are the only safety system available for removing decay heat. Their operation, however, requires dc power supplied by batteries. The redundancy and diversity of the EFW trains reduce the failure probability of secondary-side heat removal.

Improved Main Control Room Design

The System 80+ MCR design (Nuplex 80+) is an evolutionary design that is expected to provide more as well as more useful information to the operator than the System 80 design. The System 80+ MCR is still being designed. For this reason, no credit was taken in the PRA for the effect the advanced MCR on normal operations (e.g., initiating event frequency) and emergency response. See Chapter 18 of this report for the staff's evaluation of the Nuplex 80+ advanced control complex.

Normally Operating Component Cooling Water System (CCWS) and Station Service Water System (SSWS) Pumps

The CCWS is a closed-loop system that supplies cooling water flow to remove heat released from plant SSCs. Heat from the CCWS is rejected to the ultimate heat sink through the open-loop SSWS. Each of these systems (i.e., CCWS and SSWS) consists of two separate and redundant divisions. Each division has two pumps: one is normally operating, the other pump is in standby and receives a starting signal if the running pump stops. This configuration eliminates the demand failures of pumps and valves that were found to be significant contributors to risk in current-generation plants with standby CCWS and SSWS designs.

Physical Separation of Safety System Redundant Trains

Facilities are designed to provide physical separation of systems or trains of systems that perform redundant safety-related functions. This increases the availability of systems because they are protected from failures associated with internal fires, internal floods, and similar CCFs. This contributes to the reduction of risk as compared to current plant designs.

Two Diesel Generators Plus Standby Combustion Turbine-Generator (CTG)

Each of the two Class 1E ac power divisions is supplied with emergency standby power from an independent EDG. Each EDG has its own dedicated battery for starting. In addition to the two EDGs, the System 80+ design includes an onsite AAC source. This is a non-safety CTG provided to cope with SBO scenarios. The AAC source is a standby unit that is diverse and independent from the EDGs. The CTG is not normally or automatically connected to the Class 1E safety divisions. However, it can be manually aligned to supply power to either safety division via one permanent non-safety (PNS) bus when the EDGs are unavailable.

Six Vital 125-V dc Batteries With 8-Hour SBO-Coping Capability

Each of the six independent load group channels and divisions of 125-V dc vital instrumentation and control power is provided with a separate and independent Class 1E 125-V dc battery (two division and four channel batteries). Each battery is of a sufficient size to supply the continuous emergency load of its own load group for 2 hours. In addition, the batteries provide an SBO-coping capability assuming manual load shedding or the use of a load management program. This permits operating the instrumentation and control loads associated with the turbine-driven EFW pumps for 8 hours.

In-Containment Refueling Water Storage Tank

Important characteristics and functions of the IRWST are (1) has a large capacity; (2) supplies water for emergency core cooling (i.e, SI and containment spray (CS)); (3) serves as heat sink for the SDS and fission product scrubber; (4) serves as a sink for CS flow and condensate runoff, thereby eliminating the need for the recirculation mode of emergency core cooling; and (5) in conjunction with remote manual valve operation, is a source of water for flooding the reactor cavity in severe accidents. A sufficient amount of borated water is stored in the IRWST to meet all postaccident SI and CS pump operational requirements. The volume of borated water is also sufficient to flood the refueling pool during normal refueling operations. The IRWST is located at a low elevation within the containment. It eliminates the need for switching over from the injection mode to the recirculation mode during emergency core cooling operations. From a PRA point of view, this is beneficial because failures associated with the switchover in case of a LOCA are eliminated.

19.1.2.2 Evolutionary Design Features for Mitigating the Consequences of Core Damage

The following are the major features that are incorporated into the System 80+ design for the purpose of improving the capability of the containment to deal with the challenges associated with severe core-damage accidents.

Large Spherical Steel Containment

The System 80+ containment building has a larger volume and higher ultimate pressure capacity than that of most operating PWRs. The increased containment volume reduces the potential for developing detonable concentrations of hydrogen under severe-accident conditions and the potential for containment overpressure from noncondensible gas buildup. The containment pressure capacity (Service Level C value of approximately 1 MPa (145 psia) at an average steel shell temperature of 143 °C (290 °F), and estimated median ultimate containment strength of approximately 1.2 MPa (171 psia) at 143 °C (290 °F)) is sufficiently large that the containment loads associated with early challenges (e.g., hydrogen combustion and DCH) are at or below the American Society of Mechanical Engineers (ASME) Service Level C value. The high-pressure capacity combined with the increased containment volume also significantly delays the time of release associated with late containment overpressure failure challenges.

Secondary Containment Design

The System 80+ design includes a secondary containment system, consisting of a concrete containment shield building and a ventilation system to service the annulus between the containment vessel and shield building. The containment shield building is designed to provide biological shielding and protection from external missiles for the containment vessel. The AVS is an engineered safety feature (ESF) and operates after an accident to produce and maintain a negative-pressure zone in the annulus. This annulus ventilation and filtration system serves as a mechanism for substantially reducing fission-product releases after design-basis and those severe accidents in which it is operable.

In-Containment Water Storage System

The System 80+ design incorporates an in-containment water storage system, which consists of the IRWST, the holdup volume tank (HVT), the steam relief system (SRS), and the reactor CFS. In addition to the typical function of the refueling water storage tank at operating plants, this system performs water collection, delivery, and heat sink functions inside the containment during accident conditions. Containment spray water, RCS break flow, and condensed water on containment structures drain first into the HVT and eventually to the IRWST through spillways connecting the IRWST and HVT. For releases through the SRS, the IRWST serves as a suppression pool and provides for steam condensation and fission-product scrubbing. This system also supplies water for reactor cavity flooding through the CFS.

Safety Depressurization System

In addition to a core-damage-prevention function, the SDS has a mitigative function. Specifically, actuation of the SDS before core debris penetrates the vessel can reduce or eliminate the challenges associated with a high-pressure melt ejection (HPME) from the reactor vessel (such as the challenges of DCH and large hydrogen combustion events at vessel breach), thereby reducing the probability of early containment failure. Furthermore, because the discharge flow is routed through a sparger network in the IRWST and not directly into the containment atmosphere, the SDS also reduces the amount of fission products released to the containment atmosphere before a vessel breach.

Reactor Cavity Design

Specific design features have been incorporated into the System 80+ reactor cavity design to minimize the challenges posed by relevant severe-accident phenomena, including DCH, fuel-coolant interaction (FCI), and CCI. The specific reactor cavity features to deal with each challenge are summarized below.

DCH — The path from the reactor cavity to the upper containment is convoluted so that the corium is disentrained and removed from the atmosphere before it reaches the upper containment region. This design feature reduces the quantity of corium that would be dispersed into the upper compartment and, therefore, the pressure rise associated with DCH. In conjunction with the high containment pressure capacity for the System 80+ design, the cavity design serves to further reduce the probability of containment failure as a result of DCH events.

FCI — The reactor cavity is designed for 1.29 MPa (188 psid) with an American Concrete Institute calculated ultimate pressure of 1.62 MPa (235 psid). It also has a high dynamic pressure capacity, as discussed in Section 19.2 of this chapter. Furthermore, the containment structure is arranged in such a way that even if the reactor cavity wall collapses, it will not lead to a containment failure. These design features, combined with the relatively large volume of the reactor cavity and limited resistance to gas flow leaving the compartment, provide the capability to accommodate significant pressurization from quasi-static and dynamic loads, such as DCH or ex-vessel FCIs, without loss of containment integrity.

CCI — The System 80+ reactor cavity incorporates several design features that reduce the importance of CCI. These are a large cavity floor area that promotes debris spreading and increases the potential for debris coolability; a thick layer of concrete to protect the containment shell, with an additional 4.6 m (15 ft) of concrete below the liner elevation; and a manually actuated reactor cavity flood system for covering the core debris with water and maintaining long-term debris coolability. In addition, the basemat will be constructed of either limestone-common sand or limestone aggregate-type concrete because of its superior resistance to ablation compared with other commonly used basemat materials such as basaltic concrete, and because the increased production of noncondensible gas production associated with limestone-based concrete could be accommodated in the System 80+ design without adversely affecting containment failure frequency. Together, these design features significantly reduce the frequency of basemat melt-through and delay the time of melt-through during those scenarios in which this failure occurs.

Hydrogen Mitigation System (HMS)

The System 80+ design incorporates a deliberate ignition system to maintain containment hydrogen concentrations below a detonable limit. The HMS uses igniters of the glow plug design and is manually controlled remotely. The igniters can be supported by both ac and dc (plant battery via dc-to-ac inverters) supplies. Because of the proven design of the glow plug igniters proposed for use in the System 80+ design and the reliability of the electrical power sources, the HMS is expected to minimize the threat of containment failure caused by large hydrogen deflagrations or hydrogen detonations. See Sections 6.2.5 and 19.2 of this report for the staff's evaluation of the HMS.

Containment Spray System

The CSS is a safety-grade system designed to reduce containment pressure and temperature resulting from DBAs. It can also be used in a severe accident to control containment pressure and temperature and to remove fission products from the containment atmosphere and thus reduce their release to the environment. The CSS has two independent trains. The two CSS pumps take suction from the IRWST and discharge through the CSS heat exchangers and the spray header isolation valves to their respective spray nozzle headers. The spray droplets then fall to the containment floor and drain to the HVT and subsequently back to the IRWST. The spray droplets are very effective in removing energy and fission products from the containment atmosphere during sequences in which they are available. The boric acid solution of the spray water minimizes the release of iodine through iodine absorption.

Emergency Backup of Containment Sprays

An emergency containment spray backup system (ECSBS) is included in the System 80+ design to serve as an onsite pumping source independent of ac power buses, with the capability to supply water to the containment spray header from an external source when the normal CSS is not available, including during SBO events.

The ECSBS comprises the following design features: (1) a 20 cm (8-in.) diameter "tee" connection to the containment spray recirculation line; (2) an extension of 20 cm (8-in.) diameter Class 2 piping from the "tee" connection to the exterior of the nuclear annex; (3) external connections for temporary hookup of an external source of water located at or near grade; (4) a portable, onsite pumping source (e.g., fire truck) with the capability to supply sufficient flow against maximum containment pressure to maintain containment pressure below ASME Service Level C limits; and (5) prestaging of all necessary hoses, fittings, and spool pieces.

19.1.3 Safety Insights From the Internal Events Risk Analysis (Operation at Power)

These insights include (1) dominant accident sequences contributing to core damage; (2) areas where certain System 80+ evolutionary design features are the most effective in reducing risk with respect to operating reactor designs; (3) major contributors to the estimated CDF from internal events, such as hardware failures, system unavailabilities, and human errors; (4) major contributors to maintaining the "built-in" plant safety (to ensure that risk does not increase unacceptably); (5) major contributors to the uncertainty associated with the estimated CDF; and (6) sensitivity of the estimated CDF from internal events to potential biases in numerical values, to assumptions made, to lack of modeling details in certain areas, and to previously raised safety issues.

19.1.3.1 Level 1 Internal Events PRA

ABB-CE estimated the CDF for the System 80+ design to be about 2×10^{-6} per year from internal events during operation at power. In addition, CDFs for various initiating event categories were estimated and are summarized in Table 19.1. The CDFs reported for the System 80 design from which the System 80+ evolved are also shown for comparison. The total CDF for the System 80+ design was estimated to be approximately 50 times smaller than the total CDF for the System 80 design. The relative contributions (in terms of percent of total) of various initiating events to the total CDF are also shown in Table 19.1 and in Figure 19.1 for both System 80+ and System 80.

For the System 80 design, LOOP and SBO essentially dominate the CDF profile (~47 percent contribution). This is followed by LOCAs (~18 percent), the "transient" events category (~15 percent), and SGTRs (~13 percent). The contribution of ATWS sequences is relatively small (~6 percent).

For the System 80+ design, the LOCA categories of initiating events (~37 percent contribution) and the "transient" events category (~35 percent contribution) are leading contributors to the total CDF. They are followed by the SGTR initiating event (~17 percent). The contributions from LOOP/SBO (~2.4 percent) and ATWS (~3.2 percent) are relatively small.

Section 19.1.3.1.1 below presents the dominant accident sequences and the major contributors to the CDF estimates for System 80+. The design features that contribute to the reduced CDF for System 80+, as compared with System 80, are described in Section 19.1.3.1.2. Finally, the insights drawn from the uncertainty, sensitivity, and importance analyses are given in Section 19.1.3.1.3.

19.1.3.1.1 Dominant Accident Sequences Leading to Core Damage

ABB-CE identified 27 sequences initiated by internal events that contribute almost 100 percent of the estimated CDF from internal events. The top six sequences, contributing more than 80 percent of the total CDF from internal events, are summarized below.

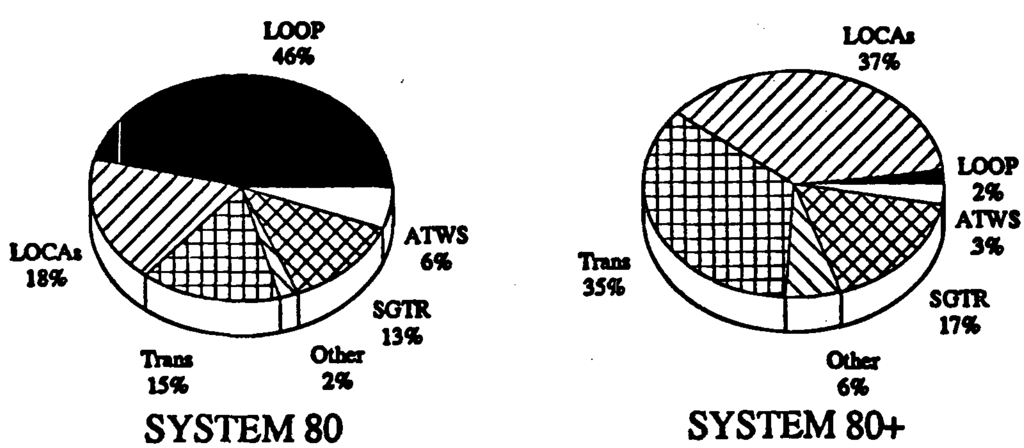
Sequence 1, with a CDF of 5×10^{-7}/year and a 27-percent contribution, is initiated by a loss of main feedwater to the SGs followed by failure of the EFWS to remove decay heat via the SGs and failure of the SDS to perform the "bleed" portion of feed-and-bleed core cooling. Risk-important failures in this sequence are CCF of check valves in the EFWS distribution or pump discharge pipe sections, CCF of the SDS bleed valves, and operator failure to initiate feed-and-bleed operation.

Sequence 2, with a CDF of 3×10^{-7}/year and an 18-percent contribution, is initiated by a medium-LOCA event followed by early or late failure of the SIS. Early SIS failure results in failure to supply makeup water and remove heat from the core. Late SIS failure results in boron crystallization, which blocks flow through the core. Risk-important failures in this sequence are CCF of SI line

Table 19.1 Comparison of contributions to CDF by initiating event

Initiating Event	System 80 (CDF/yr)	System 80+ (CDF/yr)
Large LOCA	2×10^{-6}	1×10^{-7}
Medium LOCA	4×10^{-6}	3×10^{-7}
Small LOCA	1×10^{-5}	2×10^{-7}
Steamline/Secondary Line Break (SLB)	1×10^{-6}	2×10^{-9}
Steam Generator Tube Rupture (SGTR)	1×10^{-5}	3×10^{-7}
Transients	1×10^{-5}	6×10^{-7}
Loss of Offsite Power (LOOP)	4×10^{-5}	4×10^{-8}
Anticipated Transient Without Scram (ATWS)	5×10^{-6}	5×10^{-8}
ISLOCA	5×10^{-9}	5×10^{-10}
Vessel Rupture	1×10^{-7}	1×10^{-7}
Total	8×10^{-5}	2×10^{-6}

Figure 19.1 Relative contributions to total CDF from internal events



MOVs or check valves, CCF of hot-leg check or isolation valves, and operator failure to initiate hot-leg injection to prevent boron crystallization.

Sequence 3, with a CDF of 3×10^{-7}/year and a 17-percent contribution, is initiated by an SGTR event followed by failure of the SIS to makeup and control the lost RCS inventory and inability to aggressively cool down and depressurize the RCS in order to use the low-pressure SCS to supply the necessary makeup inventory. Risk-important failures in this sequence are CCF of SI line MOVs or check valves, CCF of SIS pumps to start and run, and operator failure to perform aggressive secondary cooldown (ASC).

Sequence 4, with a CDF of 2×10^{-7}/year and a 9-percent contribution, is initiated by a small-LOCA event. The SIS fails to makeup lost RCS inventory. This is followed by failure to aggressively cool down and depressurize the RCS in order to use the low-pressure SCS to supply the necessary makeup inventory. Risk-important failures in this sequence are CCF of SI line MOVs or check valves, CCF of SIS pumps to start and run, and operator failure to perform ASC.

Sequence 5, with a CDF of 1×10^{-7}/year and a 6-percent contribution, is initiated by a large-LOCA event followed by early (DVI) or late (simultaneous hot-leg and DVI) failure of the SIS. Early SIS failure results in failure to supply makeup water and remove heat from the core. Late SIS failure results in boron crystallization, which blocks flow through the core. Risk-important failures in this sequence are CCF of SI line MOVs or check valves, CCF of hot-leg check or isolation valves, and operator failure to initiate hot-leg injection.

Sequence 6, with a CDF of 1×10^{-7}/year and a 6-percent contribution, is initiated by a vessel rupture event, that is, a breach in the primary pressure boundary that causes loss of reactor coolant in excess of the SIS capacity. This leads to core damage.

19.1.3.1.2 Risk-Important Design Features

The major design features added to the System 80+ design that contribute to the reduction in CDF as compared with System 80 (and operating reactor designs) are reported below for each of the major contributing initiating event categories.

The following are the most important features of the System 80+ design that contribute to the reduction in the estimated CDF associated with LOOP, including SBO, sequences (CDF reduced from 4×10^{-5}/year to 4×10^{-8}/year):

- separate offsite power source that bypasses the switchyard — reduces the frequency of LOOP events
- turbine-generator runback (steam bypass) capability to maintain hotel loads on a loss of grid — further reduces the frequency of LOOP events
- dedicated battery for each diesel generator — increases the reliability of the onsite Class 1E emergency ac power
- six Class 1E 125-V dc batteries with SBO-coping capability that exceeds 8 hours
- four-train EFWS (two with turbine-driven pumps) — improves reliability of secondary heat removal, which contributes significantly to the reduced risk for all sequences (with or without onsite ac power available)

The following are the most important features of the System 80+ design that contribute to the reduction in the estimated CDF associated with "transient" sequences (CDF reduced from 1×10^{-5}/year to 6×10^{-7}/year):

- larger pressurizer and SGs — reduces initiating event frequency
- four-train EFWS with redundant sources of EFW — increases the reliability of secondary heat removal, which appears in almost all sequences leading to core damage
- highly reliable, normally running CCWS and SSWS — the increased reliability of these support systems contributes significantly to increased reliability of most plant safety systems, such as SIS pumps, EFWS motor-driven pumps, and SCS pumps
- SFWS, with source from the CST — contributes to the increased reliability of heat removal through the SGs
- turbine-generator full run-back capability — reduces initiating event frequency
- two redundant and diverse EFW actuation systems — increases the reliability of secondary heat removal

The following are the most important features of the System 80+ design that contribute to the reduction in the estimated CDF associated with SGTR sequences (CDF reduced from 1×10^{-5}/year to 3×10^{-7}/year):

- Four-train EFWS — the increased reliability of this system (four instead of two or three trains) reduces the reliance on feed-and-bleed cooling as the last defense against core damage (for System 80+ the RDS can be used for feed-and-bleed cooling).
- Four-train SIS — the increased reliability of this system (four instead of two trains) reduces the importance of performing ASC for early core cooling. ASC, which is the last line of defense when SI is not available, requires use of both SGs and involves rather complicated human actions to be performed in a short time.
- SDS — provides an alternate DHR path through primary feed-and-bleed, which is much more reliable and faster than the high-pressure feed-and-bleed cooling of currently operating PWRs (replacing power-operated relief valves with MOVs simplifies operator actions and

provides flexibility for controlled and fast depressurization to SIS actuation pressures).

- Large IRWST capacity with refill capability — increases the long-term recovery probability for unisolable SG leaks, which bypass the containment, by preventing depletion of borated water and core damage.

The following are the most important features of the System 80+ design that contribute to the reduction in the estimated CDF associated with small-LOCA sequences (CDF reduced from 1×10^{-5}/year to 2×10^{-7}/year):

- IRWST — eliminates the need for the recirculation mode of emergency core cooling, which is an important risk contributor in operating PWRs.

- Four-train SIS — the increased reliability of this system (four instead of two trains) reduces the importance of performing ASC for early core cooling. ASC, which is the last line of defense when SI is not available, requires use of both SGs and involves rather complicated human actions to be performed in a short time.

- SDS — for once-through core cooling (feed-and-bleed) when all feedwater sources are unavailable.

The following are the most important features of the System 80+ design that contribute to the reduction in the estimated CDF associated with ATWS sequences (CDF reduced from 4×10^{-6}/year to 5×10^{-8}/year):

- large pressurizer — reduces frequency of transients requiring reactor scram

- large SGs — reduces frequency of transients requiring reactor scram

- SDS — allows use of the SIS pumps for long-term reactivity control when the charging pumps are unavailable

19.1.3.1.3 Insights From the Uncertainty, Importance, and Sensitivity Analyses

ABB-CE performed an uncertainty analysis to determine the magnitude of uncertainties that characterize the Level 1 PRA results (CDF from internal events) as well as the major contributors to these uncertainties. ABB-CE also performed an importance analysis to determine important contributors to risk as well as to the maintaining of the existing designed-in risk level. In addition, it conducted selected sensitivity analyses to provide insights about the impact of uncertainties (and potential lack of detailed models) on the estimated CDF and to determine the robustness of the design to biases in numerical values, such as failure probabilities, unavailabilities, and frequencies.

Insights From the Uncertainty Analysis

The System 80+ CDF estimates for internal events are reported in terms of a mean value and an associated error factor (EF). The EF is the ratio between the 95th percentile and the median (50th percentile) of the assumed log-normal distribution (which is the same as the ratio between the median and the 5th percentile). The EF is a measure of uncertainty that expresses the spread of a fitted log-normal distribution. The total CDF from internal events, as estimated by ABB-CE, has a mean value of 2×10^{-6}/year and an EF of approximately 3. Thus, the 95th and 5th percentiles are 6×10^{-6}/year and 7×10^{-7}/year, respectively. Only uncertainties associated with reliability and availability data were considered. Uncertainties associated with modeling (or lack of modeling) of accident sequences, system failure modes, and human errors were not included. The following conclusions can be reached from the results of the uncertainty analysis:

- Relatively small uncertainties are associated with the majority of the major contributors to the dominant accident sequences and total CDF.

- Relatively large uncertainties (EF higher than 10) are associated with the following major contributors to the dominant accident sequences and total CDF from internal events:

  - operator failure to perform an ASC to depressurize the RCS in order to use the low-pressure SCS to provide the necessary inventory when the SIS is unavailable during a small-LOCA or SGTR event

  - operator failure to initiate primary feed-and-bleed operation to provide an alternate DHR path when heat transfer through the SGs or the SCS is unavailable

  - CCF of all SI pumps to start

Insights From the Importance Analysis

The importance analysis performed by ABB-CE for the System 80+ design addressed two general objectives: (1) risk reduction and (2) safety or reliability assurance. The first objective (risk reduction) was achieved by the identification and ranking of dominant contributors to risk to identify areas where the plant risk can be reduced by design and/or operational changes. The second objective (reliability assurance) was achieved by the identification of dominant contributors to maintaining the designed-in risk level (to ensure that risk does not increase and is as low as the PRA indicates it is). To meet these two objectives, ABB-CE used, among others, the following two risk-importance measures to rank SSCs and human actions:

- Risk Reduction Worth: gives the factor by which the CDF decreases when an SSC or human action is assumed to be perfectly reliable (perfect component or no error); provides indication of existing margin for improvement

- Risk Achievement Worth: gives the factor by which the CDF increases when an SSC or human action is assumed not to be available or to be failed (event probability is assumed to be 1); provides indication of the importance of maintaining the existing reliability

The risk achievement worth importance measure is useful in identifying SSCs for which it is particularly important to do good maintenance, since poor reliability and availability of this equipment would increase significantly the CDF estimate. The risk reduction worth importance measure is useful in identifying SSCs that would benefit the most from improved testing and maintenance by minimizing equipment unavailability and failures.

ABB-CE performed importance analyses at both the system and component levels. Detailed results of these analyses are documented in CESSAR-DC, Section 19.15. The major insights drawn from the importance analyses are summarized below:

- Events that would decrease significantly the built-in reliability (i.e., events with highest risk achievement worth) are hardware CCFs and human errors. This is due to the redundancy and diversity of the System 80+ safety systems, which ensure that single independent hardware faults are not among those events whose occurrence would have a large impact on the CDF from internal events.

- CCF of the following sets of components was found to have a large impact on the estimated CDF from internal events (i.e., sets of components with highest risk achievement worth):

    - EDS components, such as the 125-V dc Class 1E buses, the 480-V ac load center transformers, the 4.16-kV Class 1E buses, the 480-V ac Class 1E load centers, the 480-V ac Class 1E motor control centers, the 125-V Class 1E batteries, and the EDGs

    - EFWS components, such as the distribution line check valves and the pump discharge check valves

    - SIS components, such as the SI line check valves, the SI line isolation valves, the SIS pumps, the SIS pump 4.16-kV circuit breakers, the DVI check valves, and the hot-leg injection check or isolation valves

    - SDS components, such as the RD (bleed) valves and the bleed valve power supply

    - SCS components, such as the discharge check valves and motor-operated isolation valves

- Operator failure to perform the following actions was found to have a large impact on the estimated CDF from internal events (i.e., operator actions with highest risk achievement worth):

    - initiate hot-leg injection to prevent boron crystallization during a medium or a large LOCA

    - align the CST to the EFWSTs to provide makeup water and continue DHR from the reactor core

    - initiate feed-and-bleed operation when secondary heat removal is unavailable.

- Failure of the following single components was found to have a significant impact on the estimated CDF from internal events (i.e., single components with highest risk achievement worth):

    - manual isolation valve in the line between the CST and the EFWSTs

    - check valve in the line between the CST and the EFWSTs

- Failures of components associated with the following events were found to be major contributors to the estimated CDF from internal events (i.e., events with highest risk reduction worth):

    - initiating events, such as loss of main feedwater, medium and small LOCA, and SGTR

    - CCF of the SI line check valves

    - CCF of the EFWS pump check valves or EFW distribution line check valves

    - CCF of the RD (bleed) valves

- Operator failure to perform the following actions was found to be a major contributor to the estimated CDF from internal events (i.e., operator actions with highest risk reduction worth):
  - perform ASC
  - initiate feed-and-bleed operation

As mentioned above, details on SSCs and human actions that ABB-CE found were risk significant are documented in CESSAR-DC Section 19.15. This information, which was generated by taking into account insights and assumptions from the entire PRA (i.e., all three PRA levels for both internal and external events and for all modes of operation), forms the basis for the following two lists: (1) a list of important SSCs (see CESSAR-DC Table 19.15.6-1) that the COL applicant should incorporate into the D-RAP and O-RAP (COL Action Item 19-14); and (2) a list of risk-important (critical) operator tasks (see CESSAR-DC Table 19.15.6-2) that should be taken into account in the MCR verification and validation process, as well as in the development of emergency procedures and training programs.

ABB-CE, in performing the level 1 PRA for internal events at power operation, identified the following ten critical tasks, which the operator must perform to prevent or mitigate severe accidents, that should be taken into account in the MCR design and the fixed display panel. ABB-CE makes a commitment to do this in Section 18.5.1.5.2 of the standard safety analysis report (SSAR) (Amendment Q). The process for including these tasks and the acceptability of this approach are addressed in Section 18.5.3.2.2 of this report.

- operator failure to initiate hot-leg injection (HHFFHOTLEG)
- operator failure to align the CST to EFWSTs (AHFDCST)
- operator failure to initiate feed-and-bleed (VHFFFEEDBLEED)
- operator failure to align the SCS for injection operation (JHFDRHRI)
- operator failure to align the SCS for long-term cooling (JHFDSCSLTC)
- operator failure to throttle the SIS pump in time (PHFFSIPUMP)
- operator failure to perform an ASC during an SGTR (AHFFASCSGTR)
- failure to perform an ASC during a small LOCA (AHFFASCLOCA)
- operator failure to generate safety injection actuation signal (FHFFSIAS)
- operator failure to restart the EFWS pumps and system (AHFF-RSEFW)

In designing the Nuplex 80+ MCR, it is important that no significant new human errors be introduced. To this end, during the MCR validation process, the COL applicant should qualitatively confirm that the findings from the human factors validation and verification (V&V) plan (as dispositioned) do not lead to a risk-significant increase in error potential over that represented in the System 80+ PRA human reliability assessment (HRA). If this is not confirmed, the COL applicant should model the additional risk-significant errors in an updated HRA. This aspect of the validation process is addressed in Section 8.1 of the Human Factors Engineering Verification and Validation Plan for the Nuplex 80+ (reference 3 of CESSAR-DC Section 18.4).

<u>Insights From the Sensitivity Analyses</u>

The sensitivity analyses performed by ABB-CE had the following objectives: (1) determine the sensitivity of the estimated CDF from internal events to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities; (2) determine the effect of potential lack of modeling details, such as modeling of RCP seal failures during an SBO event, on the estimated CDF from internal events; and (3) determine the sensitivity of the estimated CDF to previously raised issues, such as operator capability to perform mitigating actions outside the MCR once an accident has occurred.

The most important insights drawn from applicant's sensitivity analyses are summarized below.

- Although the estimated CDF from internal events is less sensitive to human error probabilities than CDF estimates for operating reactor designs, it is still very sensitive to operator actions that are performed outside the MCR during an accident (e.g., operator fails to align the CST to the EFWSTs for long-term cooling).
- The estimated CDF from internal events is very sensitive to several common-cause-failure probabilities. This underlines the importance of those design features

and operational requirements that aim at preventing CCFs, namely divisional separation, diversity of redundant components, as well as appropriate maintenance and training programs.

- The System 80+ CDF from internal events is not very sensitive to reasonable changes in single component failure probabilities or initiating event frequencies.

- The estimated CDF is not sensitive to further reductions in safety system outage times for test and maintenance during power operation.

- The estimated CDF from internal events is not sensitive to the RCP seal failure probability following SBO or loss-of-cooling-water event. For this reason, RCP seal LOCAs were not modeled in the System 80+ PRA. This result is due to the reduced likelihood of SBO events and the improved reliability of RCP seal cooling for System 80+ as compared to operating reactor designs. Reduced SBO likelihood is due to the following features: (1) two physically separate and electrically independent switchyards; (2) turbine-generator runback capability; (3) addition of the non-safety combustion turbine-generator, which is independent and diverse from the EDGs; and (4) EDGs with dedicated 125-V batteries. Improved reliability of RCP seal cooling is due to the redundant and diverse systems that perform this function: (1) two separate and independent CCWS and SSWS divisions, (2) two redundant and divisionally separated charging pumps, and (3) a diverse positive displacement (air-cooled) RCP seal cooling pump.

- The estimated CDF is not sensitive to the assumption, based on best- estimate calculations, that the safety injection tanks (SITs) are not required to prevent core damage during a medium LOCA.

- The System 80+ CDF from internal events is not very sensitive to increases in MOV failure rates. This result shows that the System 80+ design is not very sensitive to the concern that generic MOV failure rates (i.e., failure rates based on generic MOV failure data) may have been underestimated.

19.1.3.2 Results and Insights From the Level 2 PRA (Containment Analysis)

In the sections that follow, results and insights from the Level 2 portion of the PRA are presented. This includes the estimated probability of containment failure, a breakdown of containment failure frequency in terms of important containment failure and release modes, and finally, a summary of the risk-significant insights from the Level 2 PRA and supporting sensitivity analyses.
Comparison of the results presented in the original and the updated PRA shows that resolution of the issues raised in the DSER has resulted in substantive changes in certain portions of the analysis (e.g., the estimated probability of containment failure from certain severe-accident phenomena) and a reordering of leading contributors to risk. However, because of the robustness of the containment design and the margins between ultimate pressure capability and peak containment loads, the System 80+ design conditional CCFP and overall risk remain low.

19.1.3.2.1 Conditional Containment Failure Probability

In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity (i.e., the structural integrity definition) and (2) releases that result in significant offsite doses (i.e., the dose definition). Rather than attempt to define a "large release," the staff used the EPRI criterion of 0.25 Sv (25 rem) at 0.8 km (0.5 miles) from the reactor as the definition of containment failure in the latter case. This is judged to bound any reasonable definition of large radiological release.

The updated containment failure frequency (based on the structural integrity definition of containment failure) is about 2×10^{-7}/year. This value includes events that lead to containment basemat penetration as containment failures. This containment failure frequency represents roughly a six-fold increase from that in the original PRA. The higher failure frequency is due to an increase in the total CDF from the Level 1 analysis, combined with modeling changes that led to an increased probability of containment failure for certain severe-accident challenges. The key modeling changes involved assumptions regarding core debris coolability in a wet reactor cavity, occurrence of energetic events and their potential to fail the containment (e.g., steam explosion and hydrogen detonation), and operator actions to actuate the accident mitigation systems (e.g., CFS and HMS). The staff finds ABB-CE's modeling of severe-accident phenomena and associated containment failure modes in the updated Level 2 PRA to be comprehensive and consistent with the current understanding of these issues. However, the staff's knowledge in this area is not complete, and as a result, many of the related models and assumptions contain significant uncertainties that can affect the bottom-line numbers. Thus, caution must be used in interpreting the CCFP.

Using the structural integrity definition of containment failure, the CCFP for System 80+ is approximately

11.7 percent for internally initiated events, and 11.4 percent when internally initiated events are combined with tornado strike events. In ABB-CE's analysis, many of the containment failures are associated with containment basemat melt-through and occur well after 24 hours (e.g., ABB-CE estimates that basemat penetration would occur at 65 hours if limestone concrete was used). It is interesting to note that if such sequences are not considered to constitute containment failure, the CCFP would be approximately 4 percent (for internal plus tornado strike events). While basemat- penetration events do not represent "controlled releases" as described in SECY-90-016, these releases are into the subsoil rather than the atmosphere, and, therefore, more benign in terms of offsite radiological consequences.

On the basis of the updated PRA, the probability of exceeding a whole-body dose of 0.25 Sv (25 rem) at 0.8 km (0.5 mile) is about 5×10^{-8}/reactor-year (for internal and tornado strike events). If doses in excess of 0.25 Sv (25 rem) are considered to constitute containment failure (i.e., the dose definition of containment failure), the CCFP would be about 3 percent. The CCFP based on the dose definition is lower than that associated with the structural integrity definition because most of the structural failures (such as basemat melt-through failures) do not have significant offsite consequences.

The staff concludes that the estimated CCFP for the System 80+ design satisfies the Commission's CPG (0.10). Specifically, within the 24-hour period after core damage, which is the focus of the CPG, the probability of containment failure (using either the structural integrity or the dose definition of containment failure) is below the goal. The probability of containment failure is somewhat higher when failures beyond 24 hours are included; however, CCFP remains less than the goal using the dose definition of failure (3 percent), and is only slightly higher than the goal (11 percent) using the structural definition of failure. In SECY-90-016, the staff stated that in view of the low probability of accidents that would challenge the integrity of the containment, the CCFP for evolutionary designs should not exceed "approximately" 0.1. Furthermore, in the related SRM, the Commission directed that the CCFP objective of 0.1 should not be imposed as a requirement in and of itself. In view of the approximate nature of the CPG, the recognition that PRA results, particularly bottom-line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11-percent CCFP estimate are late, containment basemat melt-throughs rather than releases to the atmosphere, the staff concludes that the System 80+ design satisfies the Commission's CPG.

19.1.3.2.2 Leading Contributors to Containment Failure From the Level 2 PRA

The frequencies of the various containment failure modes and the fractional contributions by containment failure mode to the total containment failure frequency are presented in Figure 19.2 for both the original and updated System 80+ PRA. The updated PRA results reported here, as well as in the discussions that follow, are based on the combined frequency of internally initiated events plus tornado strike events. A separate accounting of results for internally-initiated events is not included because of the small CDF from tornado strike events (3×10^{-7}/year and the limited impact of tornado strike events on level 2 PRA results, as illustrated in Table 19.2.

The containment failure profile is significantly changed relative to the original PRA. The breakdown of results from the updated PRA reveals that 11 percent of the core-damage events involve containment failure. Most of these failures (about 70 percent) involve late containment failure. Basemat melt-through accounts for 67 percent of the containment failure frequency, compared with 6 percent in the original PRA. Late containment failures due to late containment pressurization or late hydrogen burns contribute about 3 percent, compared with less than 1 percent of the containment failure frequency in the original PRA. Early containment failures, which were a negligible contributor in the original PRA, account for about 10 percent of the containment failure frequency in the updated PRA. Containment isolation failure (primarily SGTR) and failure due to containment bypass scenarios ISLOCA contribute about 21 percent and 0.2 percent in the updated PRA, compared with 23 percent and 8 percent in the original PRA, respectively.

The leading containment failure mode in the original PRA, containment failure before core melt, is eliminated in the updated PRA because of the addition of an external spray connection and water source to the System 80+ design. The high probability of containment spray availability and the assumption that once the containment spray is available a containment overpressure failure due to steam generation is avoided result in a containment-failure-before-core-melt frequency below the 1×10^{-9} truncation level.

Important contributors to each of these failure modes are identified in Figure 19.3, and discussed further in the sections that follow.

<u>Late Containment Failure</u>

Late containment failure is defined in the System 80+ PRA as a failure more than 1 hour after reactor vessel failure. Three late containment failure modes are failure

Table 19.2 Summary of Level 2 PRA results

Containment Release Category	Fraction of Core-Damage Frequency	
	Internal Events Only[1]	Internal Plus Tornado Strike Events[2]
Intact Containment (RC1)	.88	.89
Late Containment Failure (RC2)[3]	.08	.08
Early Containment Failure (RC3)	.01	.01
Containment Isolation Failure (RC4)	.03	.02
Containment Bypass (RC5)	< .001	< .001

[1] Total CDF is estimated to be 1.7×10^{-6}/year

[2] Total CDF is estimated to be 2.0×10^{-6}/year

[3] Includes basemat melt-through and overpressure and overtemperature failures

evaluated in the System 80+ PRA. They are (1) containment basemat melt-through; (2) containment overpressurization failure due to steaming, noncondensible gas generation, or a late hydrogen burn; and (3) containment overtemperature failure.

Containment Basemat Melt-Through: This is the leading containment failure mode in the updated PRA. The frequency of basemat melt-through in the updated PRA is 1.3×10^{-7}, which is about 60 times the frequency in the original PRA. One important reason for this increase is that basemat melt-through to the underlying stone or soil was not considered a containment failure in the original PRA, but is considered as one mode of late containment failure in the updated PRA. Furthermore, although the corium was assumed coolable and CCI was assumed terminated if the reactor cavity was flooded in the original PRA, a small probability for CCI to continue in a flooded cavity was assumed in the updated PRA (1 percent). About 90 percent of the basemat melt-through events occur in sequences in which the reactor cavity is dry. Three different containment basemat melt-through modes are considered in the System 80+ PRA. They are (1) penetration of the basemat foundation, (2) erosion into and penetration of the subsphere region of the auxiliary building, and (3) erosion into the cavity floor and walls inducing cavity wall collapse, which causes reactor vessel supports to shift, and ultimately leading up to failure of one or more containment penetrations. Approximately 95 percent of basemat melt-through comes from a melt-through to the underlying stone or soil in the updated PRA. Consistent with PRAs for operating plants, the System 80+ PRA does not distinguish between melt-through of the containment shell (which is embedded about 0.9 to 1.5 m (3 to 5 ft) below the floor of the reactor cavity) and melt-through of the basemat foundation (which extends approximately 4.6 m (15 ft) below the embedded shell). Melt-through of the containment shell would occur much earlier than basemat penetration; however, the releases due to melt-through of the shell are generally considered to be negligible because of the limited flow areas and gaps (between the concrete and shell). In System 80+, all sequences that lead to basemat melt-through, by definition, involve prior melt-through of the embedded containment shell. Conversely, most sequences that lead to melt-through of the containment liner can be expected to eventually fail the basemat as well, since the core debris is not cooled in these cases. Accordingly, the frequencies of containment shell melt-through and basemat melt-through would be very similar.

Late Containment Overpressure Failure: Three mechanisms for late containment overpressure failure are considered in the System 80+ PRA. They are gradual steam pressurization with a flooded cavity and coolable corium, gradual steam and noncondensible gas pressurization with CCI in a flooded cavity or with CCI in a dry cavity, and late hydrogen burn. Results of the System 80+ PRA show that the first two mechanisms account for about 95 percent of the late containment overpressure failure frequency with roughly equal contributions from each. Late hydrogen burn contributes only 5 percent to the late containment overpressurization

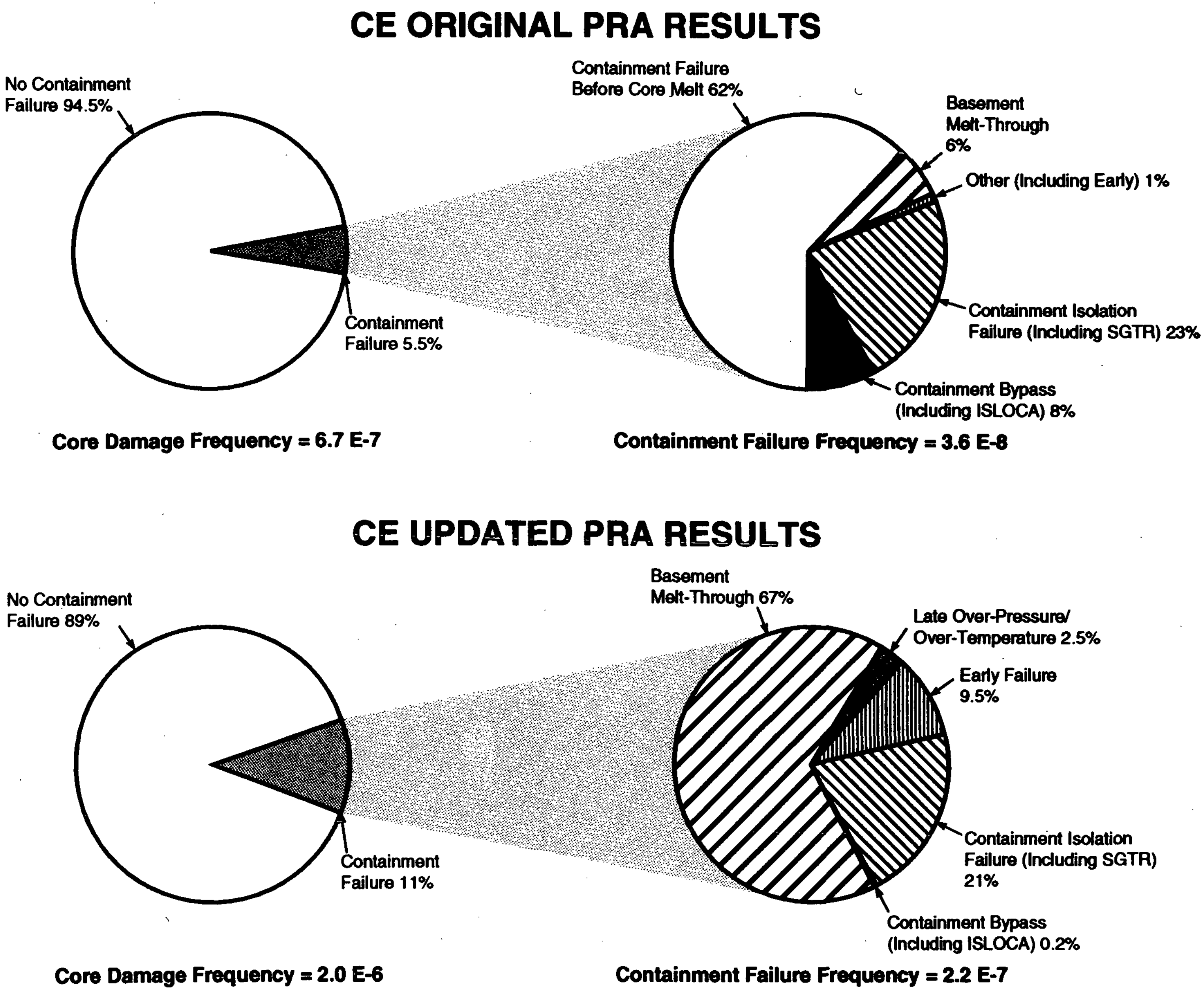
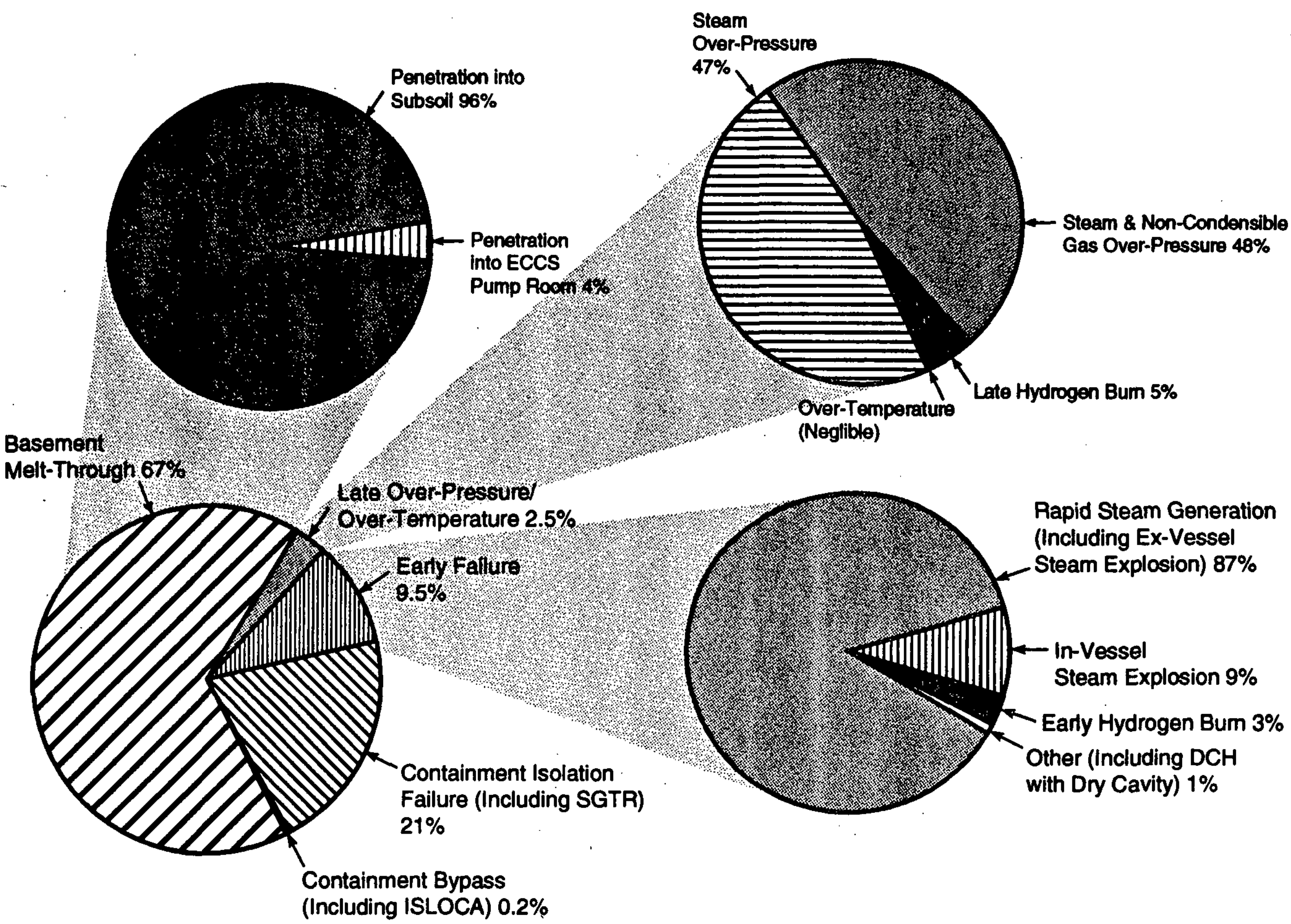


Figure 19.2 Comparison of containment failure frequency based on the original and updated Level 2 PRA results

Figure 19.3 Breakout of important contributors to containment failure based on the updated Level 2 PRA results (internal plus tornado strike events)



Containment Failure Frequency = 2.2 E-7 (CCFP=11%)

probability. The hydrogen available for a hydrogen burn in the late period includes the hydrogen produced in-vessel from zirconium oxidation and the hydrogen produced ex-vessel from CCI (including oxidation of the remaining zirconium in the melt). The challenge to the containment integrity is more severe if the hydrogen in the containment has not been consumed by early hydrogen burns. For this case, a late hydrogen burn usually involves a sudden deinerting of the containment because of the restoration of containment sprays. Because of the large containment volume and the high containment pressure capability, the challenge to containment integrity from a late hydrogen burn is not significant.

Containment Overtemperature Failure: This failure involves the loss of containment integrity as a result of a sustained high-temperature environment in the containment. Containment overtemperature failures are assumed in the PRA to be caused by high-temperature degradation of seal materials used in major containment penetrations. These penetrations consist of the equipment hatch, personnel airlocks, and the fuel transfer tube. In the PRA, containment over-temperature failure is assumed to occur only if the reactor cavity is dry and containment heat removal is lost. As discussed in Section 19.2.6.4 of this chapter, the containment atmosphere temperature under these conditions is expected to be about 204 °C (400 °F) based on ABB-CE calculations using the MAAP code, and about 260 °C (500 °F) based on staff calculations using MELCOR. The probability of a containment penetration failure under dry-cavity conditions is assumed to be 1×10^{-3} in the PRA. As a result, the contribution from overtemperature failure is negligible.

It is important to note that the detailed designs of major penetrations are not available at this time and thus were not actually analyzed by ABB-CE. ABB-CE has stated that the intent of the penetration seal design is to ensure that the selected seal and mounting will provide a minimum of 1-day containment integrity. ABB-CE has further stated that this design objective will be achieved by a combination of selecting high-quality and high-capability seals, protectively mounting the seal so that it is not directly exposed to the containment environment, and providing double seals (inner and outer) whenever possible. This design commitment is given in CESSAR-DC Section 19.11.3.1.4. The staff considers ABB-CE's design objective and the assumed failure rate in the PRA achievable. This design commitment is consistent with the results of NRC funded research into containment seal and penetration integrity under severe accident conditions. This is discussed further in Section 19.2.6.4 of this chapter.

Early Containment Failure

The total frequency of early containment failure predicted by the System 80+ PRA is 2×10^{-8}/year, or about 11 percent of the containment failure frequency. The mechanisms that are considered in the System 80+ PRA for early containment failure include in-vessel steam explosion (alpha mode failure), DCH, early hydrogen burn, rapid steam generation (RSG), rocket mode failure, and corium impingement resulting from HPME.

The major contributors to early containment failure are those from energetic events such as steam explosion and hydrogen detonation. The fractional contributions from the various containment failure mechanisms to early containment failure are 88 percent for ex-vessel FCIs, 9 percent for alpha mode failure, and 3 percent for early hydrogen burn. The contributions from DCH, rocket mode failure, and corium impingement are very small ($<$ 0.01 percent).

Ex-vessel FCIs are the leading contributor to early containment failure according to the System 80+ PRA. These events include ex-vessel steam explosion (EVSEs) and quasi-static pressurization (i.e., a pressure spike produced in the containment by the steam generated from the quenching of the high-temperature core debris). According to the System 80+ PRA, EVSE are about an order of magnitude more likely to cause a containment failure than quasi-static pressurization. The high contribution of steam explosions to early containment failure can be attributed to the high probability that the reactor cavity will be flooded using the CFS ($>$ 90 percent), the assumed probability of an EVSE occurring in a flooded cavity (0.86), and the assumed probability of containment failure given that EVSE has occurred (0.015). These parameter values control the probability of containment failure caused by FCIs and the probability of early containment failure in the System 80+ PRA.

Alpha mode failure has a frequency of 2×10^{-9}/year and accounts for about 9 percent of the early containment failure frequency. This containment failure mode involves considerable phenomenological uncertainty. The probability of an alpha mode failure depends on the RCS pressure at the time when the molten core debris (corium) contacts the water in the reactor vessel lower plenum after the failure of the core support plate. On the basis of a review of available steam explosion data and analyses, ABB-CE used a value of 1×10^{-3} for low RCS pressure and a value of 1.0×10^{-4} is used for high RCS pressure in the System 80+ PRA. These values are greater than the median values, but almost an order of magnitude smaller than the mean values of the distributions used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990. PRAs. However, even if the NUREG-1150 mean values are used to quantify the CCFP from this containment failure mode, the absolute value of containment failure frequency due to alpha mode will still be small.

Hydrogen deflagration and detonation contributes 3 percent to the early containment failure frequency. The hydrogen burn events considered in the System 80+ PRA include both deflagration and detonation. According to the System 80+ PRA, hydrogen detonation is the more important failure mechanism for early hydrogen burn. Even with the use of conservative assumptions regarding the hydrogen generation and hydrogen burn process in the PRA, the predicted CCFP for hydrogen deflagration is much lower largely because of the large containment volume and the strength of the containment. The challenge to containment integrity from a hydrogen burn is reduced in the System 80+ PRA by the use of the HMS. Because of the high reliability of the HMS and the low probability of operator error, the loss of the igniter system is dominated by the loss of power in the System 80+ PRA. Results of the sensitivity analysis show that the conditional probabilities for the various release classes (RCs) are essentially unchanged by the change in HMS failure probability.

DCH is a negligible contributor to early containment failure for System 80+ and accounts for less than 1 percent of the early failure frequency. The low frequency of DCH-induced containment failure is due to the low probability that the RCS will be at high pressure at the time of reactor vessel breach (0.05), the low probability that the reactor cavity will be dry at the time of reactor vessel failure (about 0.08), and the low probability of containment failure given the estimated loads for DCH events (0.014). It is important to note that HPME events that occur with the reactor cavity flooded are treated as RSG events rather than as DCH events in the PRA. Thus, cavity flooding has the effect of reducing the frequency of DCH failures while increasing the frequency of steam explosion failures.

The other early containment failure mechanisms evaluated in the System 80+ PRA are (1) direct shell attack via corium impingement, (2) rocket mode containment failure, and (3) reactor cavity overpressure failure. Analyses by ABB-CE showed that the probability of a reactor cavity overpressure failure is negligible and it is therefore not included in the early containment failure model in the PRA. The other two containment failure mechanisms are included in the early containment failure logic model, but their contributions to early containment failure are small and negligible.

Containment Isolation Failure

According to the System 80+ PRA, containment isolation failure accounts for about 21 percent of the total containment failure frequency in the updated PRA and is the most dominant contributor to the risk for System 80+. In the System 80+ PRA, the SGTR sequences with an unisolable path to the atmosphere outside the containment (e.g., a stuck-open atmospheric dump valve (ADV)) are considered as one mode of isolation failure. The SGTR sequences contribute about 94 percent to the total frequency of isolation failure and essentially all of its risk.

According to the System 80+ PRA, an SGTR-initiated core-damage sequence may not always lead to an isolation failure (i.e., release through the secondary system, bypassing the containment). The probability of isolation failure for an SGTR sequence depends on the core-damage scenario developed in the Level 1 analysis. For the SGTR sequences where core damage is partly caused by an unisolable leak (in the Level 1 analysis), the probability of isolation failure is taken to be 1.0; for the SGTR sequences where core damage is not affected by the secondary-side isolation status (in the Level 1 analysis), an isolation failure probability of 4 percent is used in the PRA for Level 2 analysis. This 4-percent isolation failure probability is based on an assessment of the failure probabilities of the ADVs, main steam safety valves (MSSVs), main steam isolation valve (MSIV), and turbine bypass and stop valves on the secondary side and has a significant effect on the total risk. Accordingly, these components have been included in the list of items to be addressed in the COL applicant's RAP.

An SGTR release can also occur if some of the SG tubes experience a high-temperature creep failure during the core-damage process of a high-pressure sequence. This type of SGTR release was not treated in the System 80+ PRA because of a rough estimate performed by ABB-CE that shows that the frequency and consequences associated with this release category are low. The low frequency is primarily due to the high probability of RCS depressurization and the low probability of a temperature-induced tube rupture given that the RCS pressure is high (at about the pressurizer safety valve setpoint). The low consequences are due to the expected short duration for fission-product release through the ruptured SG tubes (i.e., the time between tube rupture and vessel breach). The release through the secondary side is expected to be terminated most likely after vessel breach because of the RD of the RCS and the subsequent depressurization of the SG and reseat of the MSSVs after vessel breach. The staff estimates that if 2 percent of all high-pressure core-damage sequences result in a temperature-induced SGTR (the failure rate used in NUREG-1150 analyses), the total frequency of SGTR sequences (isolated plus unisolated) would increase by about 5 percent. Even if all of the temperature-induced SGTR events were assumed to be unisolated, the frequency of containment isolation failure (currently 21 percent of all containment failures) would increase by about one-third. The staff finds ABB-CE's treatment to be adequate given the relatively small impact on total risk.

The contribution to the containment isolation failure frequency from traditional failure mechanisms (i.e., failure to successfully isolate containment penetrations), although modeled using a failure probability equivalent to that in PRAs for operating reactors, is relatively small for System 80+.

Containment Bypass

In the System 80+ PRA, the only accident sequences that are classified as containment bypass events are the ISLOCA sequences. These sequences traditionally are a major concern because the environmental release generally begins early and does not receive the benefit of any containment holdup and fission-product retention. However, as a result of piping system upgrades discussed previously, the frequency for ISLOCA sequences is very low for System 80+ (5×10^{-10}/year). As such, these sequences contribute only 0.2 percent to the total containment failure frequency and about 1 percent to the total risk.

The containment bypass RC is characterized in the PRA by a failure of the check and isolation valves in one SCS line resulting in a catastrophic failure of this line outside the containment. The associated release path is through the broken SCS line into the subsphere region of the auxiliary building. A high decontamination factor is used in the

PRA for these sequences because of both the long release path and the potential for water scrubbing. However, because the containment bypass frequency for System 80+ is so low, its contribution to total risk is not expected to become significant even if a lower decontamination factor is used in the analysis.

19.1.3.2.3 Important Insights From Level 2 PRA and Supporting Sensitivity Analyses

Insights from the Level 2 PRA are summarized below. These are organized in terms of equipment and design features, severe-accident phenomena and challenges, and human actions.

Equipment/Design Features

The breakdown of the core damage events that involve containment failure reveals that the bulk of these failures (70 percent) involve late containment failure, primarily due to a basemat melt-through. An additional 10 percent of the containment failure frequency comes from early containment failure, including failures caused by RSG or explosion, DCH, and hydrogen combustion. Another 21 percent comes from containment isolation failure, which in the updated PRA includes SGTR failure with un-isolated secondary system. Containment bypass failure contributes about 0.2 percent.

The System 80+ containment building offers several benefits with regard to severe accidents, specifically: (1) the increased containment volume reduces the potential for developing detonable concentrations of hydrogen under severe accident conditions, and essentially eliminates containment overpressure from noncondensible gas buildup as a major contributor to containment failure, and (2) the containment pressure capacity and estimated median ultimate containment strength are sufficiently large that the containment loads associated with early challenges, for example, hydrogen combustion and DCH, are at or below the Service Level C value. This assures a very low probability of containment failure for such challenges. The high-pressure capacity, combined with the increased containment volume, also delays significantly the time of release for late containment overpressure failure challenges.

The high pressure capacity of the reactor cavity, combined with the ability to maintain containment structural integrity even if a wall collapses in the reactor cavity, provides the capability to accommodate significant pressurization from quasi-static and dynamic loads, such as DCH or ex-vessel FCIs without loss of containment integrity. As a result, the conditional probability of containment failure due to ex-vessel FCIs is very small (on the order of 1 percent). Although FCIs still account for a significant fraction of the containment failures, the capability of the reactor cavity to withstand such loads assures that the frequency of these failures is low in absolute terms.

Hydrogen combustion is not a significant contributor to containment failure, even if the HMS is unavailable. A sensitivity analysis shows that complete unavailability of the system would result in a negligible change in the RC frequencies. The ability of the containment to accommodate hydrogen combustion without the igniter system is due in part to the large volume and high ultimate pressure capacity of the System 80+ containment.

The ECSBS is an important feature for assuring a low probability of late containment failure. The updated System 80+ PRA shows that the availability of this external source of water to the sprays virtually assures containment heat removal in the long term. Consequently, the dominant containment failure mode in the original System 80+ PRA, "Containment Fails Before Core Melt," is eliminated in the updated PRA. Sensitivity analyses using the updated PRA show that unavailability of the backup system would increase CCFP from 11 percent to about 14 percent. The COL applicant will submit a detailed system design, and will determine specific pump flow rates and the location of all associated valves and connections; this is COL Action Items 19-10 and 19-11. The COL applicant will develop detailed procedures for use of the system as part of COL Action Item 19-16.

The CFS is an important feature for assuring a low probability of late containment failure, that is, basemat penetration. Even though basemat penetration constitutes the major contributor to containment failure, about 90 percent of the basemat melt-through events occur in sequences in which the reactor cavity is dry. Sensitivity analyses show that the probability of late containment failure would increase from 8 percent to about 53 percent if the probability of successfully flooding the cavity is reduced by a factor of 2. Thus, it is important to assure that the reactor cavity is filled with water in the long term. The COL applicant would develop specific details regarding actuation time and the parameters that will be used to determine the need for cavity flooding as part of its plant-specific severe-accident management guidelines. This is COL Action Item 19-16. The reliability of the CFS and associated valves is important and should be monitored by the COL applicant's reliability assurance program.

The type of concrete selected for use in the containment basemat (e.g., limestone or basaltic) does not have a significant impact on either containment failure frequency, or overall plant risk.

Phenomena/Challenges

Containment performance is not strongly affected by RCS pressure at the time of vessel breach. The PRA shows that containment failure is dominated by ex-vessel FCIs which occur in a flooded cavity (the reactor cavity is expected to be flooded via the CFS in more than 90 percent of the sequences). The probability of FCI-induced containment failure is assumed in the PRA to be independent of whether vessel breach occurs at high or low pressure. Although depressurization would eliminate the potential for DCH, this is only a very small contributor to containment failure for System 80+. Furthermore, the probability of early containment failure from FCI assumed in the PRA is approximately equal to probability of containment failure due to DCH. As a result, the frequency of early containment failure is not significantly impacted by assumptions regarding operator actions to depressurize or creep rupture of RCS piping.

The PRA shows that risk reductions are possible through refinement of the reactor cavity flooding strategy, which presently calls for flooding the cavity preceding reactor vessel breach. Specifically, for sequences with low RCS pressure at vessel breach, delaying reactor cavity flooding until after reactor vessel failure would eliminate the potential for significant ex-vessel FCIs coincident with vessel failure. Subsequent cavity flooding would result in either a coolable debris bed in one extreme, or continued CCI and eventual late basemat melt-through in the other extreme. However, in either case, the associated consequences would be more benign than an early containment failure. An alternative to delayed flooding would be to fill the cavity only partially preceding vessel breach, and to complete the flood-up after the vessel breach. This would reduce the height of water over which dynamic loads would be transmitted to the cavity walls, while retaining the benefit of quenching of the core early. For high-pressure sequences, the condition of the cavity at vessel breach does not significantly affect the probability of early containment failure since the probability of containment failure from DCH in a dry cavity is approximately equal to the probability of containment failure from ex-vessel FCIs in a flooded cavity. Thus, the same reactor cavity flooding can be used regardless of the RCS pressure. As discussed below, specific details regarding system actuation will be addressed in the severe accident management guidance and procedures to be developed by the COL applicant. This is COL Action Item 19-16.

Human Response

Operator actions to isolate a faulted SG have a major impact on the estimated frequency of containment isolation failure (RC4). Increasing the probability of failure to isolate from 4 percent to 20 percent increases by a factor of 2 the conditional probability of containment isolation failure (from 0.023 to 0.047). This increase is significant in terms of offsite consequences since the probability of large releases for System 80+ is dominated by this release category. Operator actions to isolate a faulted SG are one of several actions identified for further assessment by the COL applicant, as discussed below.

Operator actions are required for actuation of the severe-accident-related features in the System 80+ design, including (1) the RDS, (2) the ECSBS, (3) the reactor cavity flood system, and (4) the HMS. Basic assumptions were made in the PRA regarding how these systems would be utilized, for example, it was assumed that the CFS would be actuated early enough that the cavity would be flooded before reactor vessel breach. Detailed procedures for use of the severe accident design features will be developed by the COL applicant, as part of COL Action Item 19-16.

Use of the ECSBS requires multiple operator actions outside the MCR. According to the basic event probability calculations in the PRA, these actions include dispatching a crew to the subsphere region, local alignment of certain valves, transport of the pumping device to the appropriate location, connection of the pumping device to the containment spray line, running the hose to the cooling pond, and starting (and monitoring operation of) the pumping device. In developing the detailed system design, including the location of all associated valves and connections, the COL applicant should take into account expected radiation levels and shielding requirements for any required local operator actions. Procedures for using the system are assumed to exist in the PRA, and will be developed by the COL applicant as part of the accident management plan. This is COL Action Item 19-15.

Risk significant operator actions were identified from the PRA based on sensitivity/importance analyses, supplemented by engineering judgment. ABB-CE judged that the following operator actions in the Level 2 analysis were outside the scope of the PRA/HRA critical task criterion, but significant enough that they should be incorporated in the functional task analysis to be performed as part of the detailed MCR design, and that the resulting indication and control requirements should be incorporated in the availability verification activity:

- Align CVCS to fill the IRWST following SGTR and following containment breach (UHFDRFIRWSTSGTR).
- Reclose the ADVs on the ruptured SG

(DHFFRECLOSEADV).

- Initiate the CFS (including during SBO events) (GHFFCFSMOVS).
- Connect the ECSBS to the spray header (GHFFECSBS).
- Align the SCS pump for backup to the containment spray pump (JHFLSIXCON1).
- Depressurize the RCS preceding reactor vessel breach (NOSDSDP).
- Isolate LOCAs outside containment at the containment interface (OIC).
- Energize the HMS (OPIGNITOFF).

This commitment is recorded as Item 107 in the human factors open issue tracking system, and is discussed further in Section 18.8.3.2.3 of this report. The activities under which this will be carried out are discussed further in Chapter 18 of this report.

19.1.3.3 Results and Insights From the Level 3 PRA (Offsite Consequences)

In the updated System 80+ PRA, the end-states of the containment event trees were grouped into 23 individual RCs. For each class, the timing and energy of release were established based on plant-specific thermal-hydraulic calculations using the MAAP code, and the isotopic content and magnitude of release were estimated using a version of the NRC-developed XSOR code, modified to reflect System 80+ design features. The NRC-developed MACCS code was then used to calculate offsite consequences for each of the RC, specifically, the whole-body dose complementary cumulative distribution function (CCDF) at 0.8 km (0.5 mile) from the reactor site, and the total person-rem exposure over a 80 km (50 mile) radius from the plant. These analyses were supplemented by sensitivity analyses to assess the impact of uncertainties in key parameters. The staff finds this overall approach and the use of the above codes to be consistent with the present state of knowledge regarding severe accident modeling and, therefore, acceptable.

In the sections that follow, results and insights from the Level 3 portion of the PRA are presented. This includes the estimated probability of exceeding selected dose criteria, a breakdown of the total risk in terms of important RCs, and finally, a summary of the risk-significant insights from the Level 3 PRA and supporting sensitivity analyses.

19.1.3.3.1 Offsite Consequences for System 80+

On the basis of the updated PRA, the probability of exceeding a whole-body dose of 25 rem at 0.5 mile is about 5.3×10^{-8} per reactor-year (for internal and tornado strike events). This value is approximately 20 times lower than the 1.0×10^{-6} goal established by the EPRI in the Advanced Light-Water Reactor (ALWR) Utility Requirements Document (URD). It should be noted, however, that the EPRI goal applies to both internal and external events, and that the results for System 80+ do not include the contribution from seismic and fire events.

The total person-rem exposure over a 80 km (50 mile) radius from the reactor is estimated to be approximately 17 person-rem over a 60-year plant life. This is based on the use of population and weather data developed by EPRI to bound 80 percent of the reactor sites in the United States. This risk is very low compared to the current generation of operating plants and is due to a combination of three factors: (1) the low estimated CDF for System 80+, (2) a low CCFP, and (3) the late and generally benign nature of releases predicted for the majority (about 70 percent) of the containment failures.

19.1.3.3.2 Leading Contributors to Risk From Level 3 PRA

The consequences calculated for each of the 23 individual RC were weighted on the basis of their individual frequencies, and combined into 5 general RCs, representing the following types of containment response:

- intact containment — RC1
- late containment failure — RC2
- early containment failure — RC3
- containment isolation failure (including unisolated SGTR sequences) — RC4
- containment bypass (including ISLOCA) — RC5

The contribution to risk from each of these 5 general RCs is presented in Table 19.3 and Figure 19.4. The following can be noted:

- Events in which the containment remains intact (RC1) account for nearly 90 percent of core damage events, but are negligible contributors to risk.
- Late containment failures, although contributing about 70 percent of the containment failure frequency, account for less than 5 percent of the risk. This is due

Table 19.3 Fractional contribution to offsite consequences from major RCs

Containment Release Category	Fractional Contribution	
	Probability of Exceeding 25 Rem At 0.5 Mile[1]	Total Person-Rem[2]
Intact Containment (RC1)	0	< .001
Late Containment Failure (RC2)	.04	.02
Early Containment Failure (RC3)	.12	.10
Containment Isolation Failure (RC4)	.83	.87
Containment Bypass (RC5)	.01	.01

[1] Overall frequency of exceeding 0.25 Sv (25 rem) at 0.8 km (0.5 mile) is estimated to be 5.3 x 10^{-8}/year

[2] Total person-rem is estimated to be 17 person-rem over a 60-year plant life, based on release class frequencies reported in Table 19.12.3-1 of ABB-CE letter OPS-93-0934 (November 1, 1993), and estimated mean doses per event reported in ABB-CE letter LD-93-143 (September 30, 1993)

to the benign nature of the bulk of these releases (i.e., basemat melt-through), and the late time of release.

- Although containment isolation failures (RC4) account for about 20 percent of the containment failure frequency, these releases dominate risk in terms of both the probability of exceeding 25 rem and the total person-rem exposure. This is because these releases do not receive the benefit of fission product holdup and scrubbing in containment.

- Releases from containment bypass events (RC5), although generally equivalent to isolation failures (RC4) in terms of the magnitude of release, account for only 1 percent of the total risk. This is due to the low estimated frequency of bypass events, which is about 2 orders of magnitude less than the frequency of containment isolation failures.

Essentially all of the containment isolation failures in RC4 involve SGTR events. About 70 percent of the frequency and risk of these SGTR events comes from RC4.36, which is characterized as an SGTR with successful injection and successful operation of the EFWS. However, RCS pressure control is not established and the ruptured SG is not isolated. Leakage to the secondary side remains high, and the operator fails to replenish the IRWST inventory. This leads to IRWST depletion and core damage at approximately 25 hours.

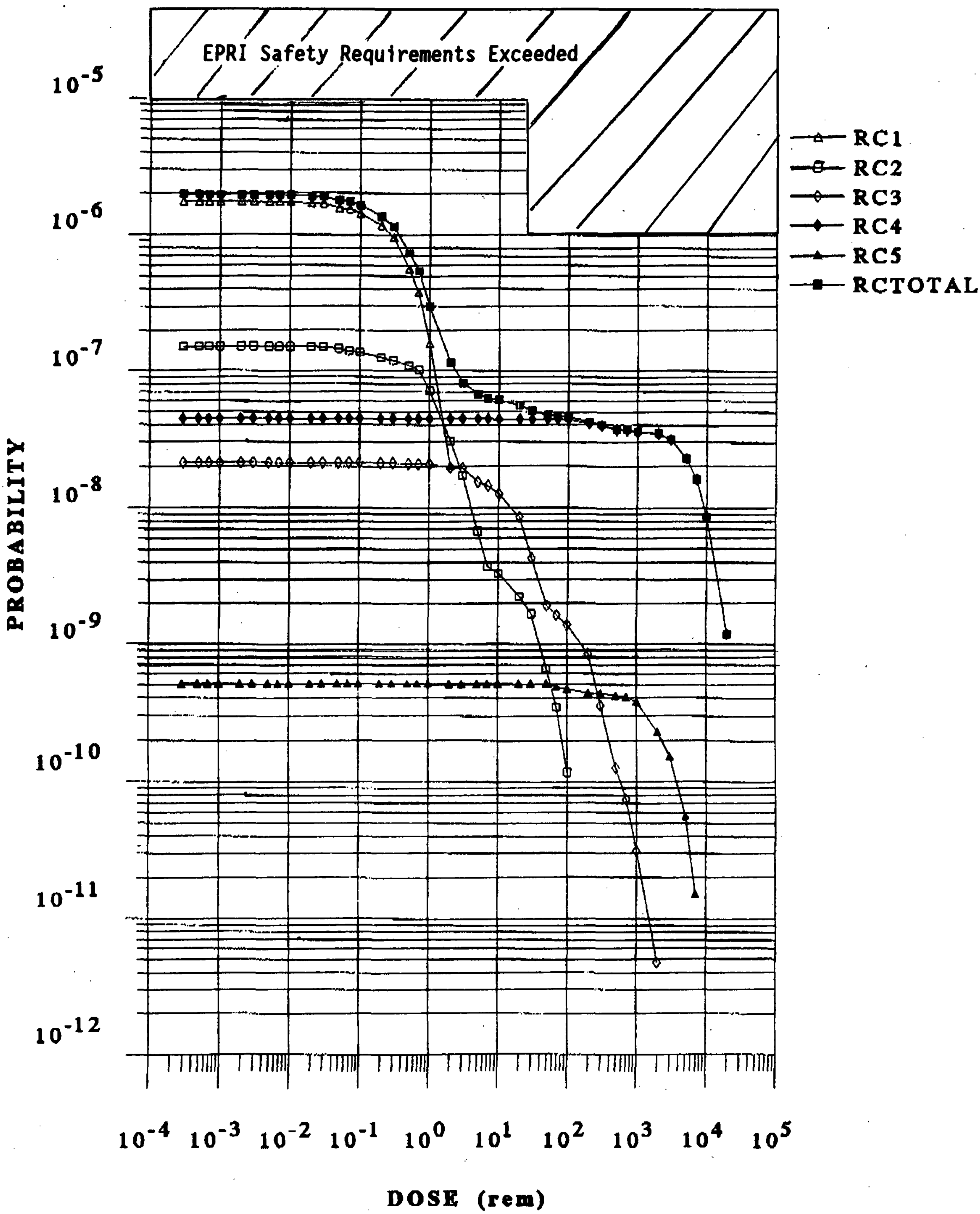
19.1.3.3.3 Important Insights From Level 3 PRA and Supporting Sensitivity Analyses

Insights from the Level 3 PRA are summarized below based on the Level 3 PRA results and supporting sensitivity analyses.

- On the basis of the updated PRA, the probability of exceeding a whole- body dose of 0.25 Sv (25 rem) at 0.8 km (0.5 mile) is about 5.3 × 10^{-8}/reactor-year (for internal and tornado strike events). If doses in excess of 0.25 Sv (25 rem) are considered to constitute containment failure (i.e., a dose definition of containment failure), then the CCFP would be 2.7 percent.

- On the basis of the updated PRA, most of the risk comes from SGTR events in which RCS pressure control is not established, the ruptured SG is not isolated, and the operator fails to replenish the IRWST inventory. ABB-CE has specifically identified operator actions to isolate the faulted SG as a "critical task" as discussed earlier. As such, these actions will receive increased emphasis by the COL applicant as part of the detailed MCR design process and the development of plant-specific operating procedures and training programs. ABB-CE has also identified operator actions to align CVCS to fill the IRWST (following an SGTR with containment breach) as important enough that they be included in the functional task analysis to be

Figure 19.4 CCDF of whole body dose at 0.5 mile by RCs (internal plus tornado strike events only)



performed by the COL applicant as part of the detailed MCR design, and that the resulting indication and control requirements be incorporated in the availability verification activity.

- ISLOCAs are not significant contributors to overall plant risk, primarily due to a piping upgrade that led to a low estimated frequency of these events. However, even if the frequency of ISLOCA is increased by 2 orders of magnitude, the overall risk would increase by only a factor of 2.

- Because of the relatively small contribution to risk from basemat melt-through events, overall plant risk is not significantly impacted by increases in the frequency of basemat melt-through (e.g., a factor of 2)

- The concrete ablation rate in basemat melt-through sequences has only a minor impact on overall risk for System 80+. A sensitivity calculation in which the time of fission product release for basemat melt-through was decreased from 65 hours to 30 hours, and the fraction of the basemat melt-throughs into the SIS pump room was doubled (from 4 percent to 10 percent) shows that risk would increase by about 20 percent. This indicates that the use of either limestone or basaltic concrete in the basemat would not have a significant impact on risk for System 80+.

19.1.4 Safety Insights From the External Events Risk Analysis (Operation at Power)

Four external events were analyzed in the System 80+ PRA: seismic events, internal fires, internal floods, and tornados. In many PRAs performed to date, these external events have had combined CDFs that are the same magnitude as for internal events. It is not unusual to see the combined CDFs for these events in the 1E-4 per year range. The methods used in the System 80+ PRA to evaluate external events is acceptable to the NRC because they provide the insights necessary to determine if any design or procedural vulnerabilities exist for these external events and because the methods provide insights needed for design certification requirements, such as ITAACs.

In SECY-93-087, the NRC identified the need for a site-specific probabilistic safety analysis and analysis of external events. ABB-CE did not perform an analysis (PRA or bounding) of the capability of the System 80+ design to withstand external flooding, hurricanes, or other site-specific external events. ABB-CE did submit evaluations of seismic, tornado, fire, and internal flood events. The NRC requires, where applicable to the site, that the COL applicant perform a site-specific PRA-based analysis of external flooding, hurricanes, or other external events pertinent to the site to search for site-specific vulnerabilities. This is COL Action Item 19-12. In addition, the site-specific PRA should update the System 80+ PRA to account for the detailed design of the as-built plant, with special emphasis on those areas of the design that either were not part of the certified design or were not detailed in the certification. The site-specific PRA should be submitted at the time of the COL application and updated, as necessary, to account for ongoing first-of-a-kind engineering. This is COL Action Item 19.1.4-1.

19.1.4.1 PRA-Based Seismic Margins Analysis

System 80+ is designed to withstand a 0.3 g SSE. Since the analyses used in designing the capability of structures, systems and components (SSCs) to withstand the SSE have significant margin in them, it is expected that a plant built to withstand the SSE actually will be able to withstand a much larger earthquake. A PRA-based margins analysis systematically evaluates the capability of the designed plant to withstand earthquakes without suffering core damage, but does not estimate the CDF from seismic events. The margins analysis is simply a reasonably conservative method for estimating the "margin" above the SSE, that is, how much larger than the SSE an earthquake must be before it compromises the safety of the plant.

The capability of a particular SSC to withstand beyond-design-bases earthquakes is measured by the value of the peak ground acceleration (g-level) at which there is a HCLPF. The HCLPF capacity of a certain SSC corresponds to the earthquake level at which, with high confidence (95 percent), it is unlikely (probability less than 5×10^{-2}) that the SSC will fail. An HCLPF value for the entire plant is determined by finding the lowest sequence HCLPF that leads to core damage. It is a measure of the capability of the plant to withstand beyond-design-basis earthquakes without suffering in core damage. The plant HCLPF value, which is assessed from the SSC HCLPF values, has units of acceleration. The NRC has indicated (SECY-93-087) that it expects that a plant truly designed to withstand a 0.3 g SSE should have a plant HCLPF at least 1.67 times the SSE (i.e., 0.5 g). The PRA-based SMA shows that the System 80+ design meets (and exceeds) the 0.5 g HCLPF value expectation.

19.1.4.1.1 Dominant Accident Sequences (Seismic)

The event and fault trees developed for the internal events PRA were modified to accommodate seismic events. In this way the random failures and human errors modeled in the internal events portion of the PRA are captured in the seismic analysis. The underlying assumption that earthquakes exceeding the SSE will happen less than once in a thousand years was used to exclude random failures or human errors with probability less than 1×10^{-3} (1×10^{-2} if they affect only one redundant train in only one safety system). This is because the combination of seismic events having acceleration higher than 0.5 g with random failures lower than 1×10^{-3} would result in estimates of CDF much less than 1×10^{-6}/year. The modified event and fault trees were merged, and cutsets were generated for all sequences that lead to core damage. These cutsets are of two kinds. One kind contains only seismic failures (i.e., without any random failures or human errors). The other kind contains random failures and/or human errors (truncated at 1×10^{-3}) in addition to seismic failures. In "quantifying" these cutsets, the HCLPF values of the seismic events (instead of mean values of failure probabilities) were used, while the probabilities of random failures and human errors are the same as for the internal events PRA. ABB-CE used the conservative deterministic failure margin (CDFM) approach for calculating HCLPF values for components and structures and the minimum/maximum approach[1] for the sequence and plant-level HCLPF calculations. The CDFM method is a set of guidelines (e.g., how to modify ground response spectra, damping, material strength, and ductility) that if followed will result in an acceptable estimate of an SSC's HCLPF. The staff reviewed these calculations, and the staff finds the calculations were properly conducted and the employed methodologies were properly applied.

ABB-CE used an initiator event tree to access seismically induced initiator HCLPF values, which were then transferred to the appropriate event tree for that initiator. The initiators considered were LOCAs, ATWS, LOOP, and general plant transients. In implementing the initiator event tree, ABB-CE assumed that each of the above initiators was mutually exclusive. This is not actually true, since it is possible to have ATWS and/or LOOP in combination with a LOCA (or even with each other). In fact, LOOP (as specified in both the EPRI and NRC margins guidance) should be assumed in all cases (because its HCLPF value is so low). Further, the staff believes that ATWS should be assumed to occur whenever a seismically induced LOCA occurs (since the ATWS HCLPF is much lower than the LOCA HCLPF). The NRC review has investigated the effect of correcting these errors and has concluded that their correction has no effect on the results or insights from the margins assessment.

The PRA-based SMA, performed by ABB-CE and reviewed by the staff, identified 31 sequences that lead to core damage (for details see CESSAR-DC Section 19.7.5). The NRC has identified those sequences that are the "dominant" contributors to the plant HCLPF value. The word "dominant" appears in quotes to emphasize that the use of this terminology in the context of a seismic margins study should not be taken in the same way as for a conventional PRA. Although these sequences (and associated cut sets) dominate the HCLPF values for the plant, the margins approach does not permit a determination that these are the dominant contributors to seismic risk in a probabilistic sense. If random failures and human errors are ignored (i.e., when cutsets containing seismic failures only are considered), the plant HCLPF was estimated to be 0.73 g. Since the plant HCLPF can be lower when certain random failures (or human errors) occur simultaneously with the seismic failure of certain SSCs, cutsets containing both seismic and non-seismic failures were examined to find out if there were any cutsets which would lower the plant HCLPF below 0.73 g or even below 0.5 g. The most significant cutset was the product of two events: (1) seismic failure of the alternate onsite ac power source (which has an HCLPF value of 0.36 g) and (2) random failure of both EDGs (which has a probability of about 1×10^{-2}).

For earthquakes that generate higher accelerations than the plant HCLPF, we no longer have the same high degree of confidence that the core will not be damaged. However, it is unlikely that a cliff effect exists for the System 80+ design (or for any other design or plant for that matter) at or near the plant HCLPF. Therefore, it is expected that the plant will have some margin (perhaps quite a bit) above the HCLPF value.

Two "dominant" seismic core damage sequences were identified by the "seismic margins" analysis. One has the lowest HCLPF (when cutsets with seismic only failures are considered), the other has the lowest combination of HCLPF with random failure/human error (when cutsets with both seismic and non-seismic failures are considered).

Sequence 1, with an HCLPF value of 0.73 g, is a seismically induced gross structural failure of the containment. The dominant failure modes for the seismic failure of the containment were found to be vertical rotation/ overturning of the containment shell with the pedestal or vertical rotation of the interior containment structures within the containment shell. These failure modes are assumed to result in severing of the containment penetrations and a transient which cannot be mitigated,

[1] In the minimum/maximum approach, if there is an "ORed" sequence where the failure of any individual SSC would cause core damage, the staff takes the lowest individual SSC HCLPF. If there is an "ANDed" sequence where the failure of all SSCs would cause core damage, the staff takes the highest individual SSC HCLPF as the sequence HCLPF.

thus ending up in core damage and loss of containment integrity. Sequence 2 is a SBO sequence. This sequence is important when cutsets with both seismic and non-seismic failures are considered. The most important cutset, associated with this sequence, starts with a seismically induced LOOP followed by non-seismic (random) failure of both EDGs and seismic failure of the standby combustion turbine. The station batteries (with load management) can provide sufficient instrumentation and control power to allow the turbine-driven EFW pumps to remove decay heat for over 8 hours. However, offsite power cannot be restored for at least 24 hours. This causes the batteries to be depleted and the turbine-driven EFW pumps to fail. Core damage is assumed to occur soon thereafter. The HCLPF for this cutset is (0.36 g) (1×10^{-2}), the product of the combustion turbine HCLPF value and the non-seismic failure probability of both EDGs).

The NRC has focused on the performance of the insights drawn from the System 80+ SMA. Because of the larger uncertainty in hazard curves and because seismic PRA results are dominated by the tails of the site hazard and SSC fragility curves, it is expected that if a seismic PRA had been performed, it would have been one of the largest contributors to CDF (though still a low absolute value). This is particularly true for the System 80+ design since the estimated CDF from internal events has been driven down so far by design changes/improvements. As it is, the System 80+ design should be better able to resist high seismic events than most, if not all, existing nuclear power plants east of the Rocky Mountains because of its built-in margins.

ABB-CE's analysis determined sequences leading to core damage only, and did not separate the results into plant damage states (PDSs). Thus, ABB-CE did not perform an evaluation to see how the containment would perform under high g-levels. That is, no specific evaluation was performed to identify paths by which the containment could be bypassed, fail to isolate, or fail. Since the plant HCLPF is in excess of 0.5 g, this is acceptable to the NRC (SECY-93-087).

19.1.4.1.2 Risk-Important Features and Operator Actions (Seismic)

The margins approach does not allow a determination of which plant features are most important to risk using importance analyses. The margins approach does allow one to determine which plant features are important to the plant-level HCLPF and the redundancy/diversity available in achieving that HCLPF. In order to make this determination, the NRC examined each sequence that leads to core damage on the seismic event trees. None of the sequences has a seismic-only HCLPF less than 0.5 g. The sequences were examined to determine if lowering the HCLPF value of a single SSC (to a much lower HCLPF value) or increasing the demand failure rate of a single system (to a much higher demand failure rate) would result in a plant HCLPF less than 0.5 g.

<u>Cutsets Containing Seismic Failures Only</u>

With regard to the seismic-only cut sets, a review of the "dominant" accident sequences (reported in Section 19.1.4.1.2 above) shows that most cutsets contain at least two seismic failures of SSCs with HCLPF values above 0.5 g. However, a more detailed review of these sequences, as well as the review of cutsets for additional sequences, identified two classes of cutsets for which this is not true. The first class includes cutsets with only one seismic failure (i.e., only one seismic failure is required for core damage to occur) and the plant-level HCLPF could be completely controlled by that event. The second class includes cutsets with multiple seismic failures (i.e., two or more seismic failures are required for core damage to occur) but only one of these events has a HCLPF above 0.5 g. If the value of this event is reduced below 0.5 g, the plant level HCLPF could also be reduced below 0.5 g but not below the value of the next highest HCLPF in the cutset.

Areas of seismic-only cutsets of the first class, that is, with only one seismic failure, are discussed below[2].

- <u>Structural integrity</u> — There are a number of important safety-related structures whose seismically induced failure would lead directly to core damage. These include the nuclear annex building (1.1 g), the nuclear island structure (0.80 g), shield building (1.25 g), containment structure (0.73 g), reactor vessel supports (1.14 g), SG supports (0.87 g), and RCP supports (0.86 g). The SMA assumes that these structures will all have HCLPF values in excess of 0.7 g. If any of these structures were built with an HCLPF lower than 0.5 g, the plant HCLPF would also be lower than 0.5 g.

[2] Note that, as discussed above, the NRC considers LOOP to result from a seismic event. Therefore, the occurrence of LOOP in a cut set is not considered to be a "second event."

- Class 1E EDS — There are a number of important safety-related electrical components whose seismically induced failure would lead directly to core damage. These include the 4.16-kV ESF switchgear (0.95 g), the 125-V dc Class 1E distribution panels (1.06 g), and the 125-V dc Class 1E circuit breakers (1.06 g). If any of these components were installed with an HCLPF lower than 0.5 g, the plant HCLPF would also be lower than 0.5 g.

Areas of seismic-only cutsets of the second class, that is, with multiple seismic failure events, are discussed below. In the System 80+ design, all of these cases involve the failure of the backup CTG (0.36 g) in combination with one other event whose HCLPF is greater than 0.5 g.

- Structural integrity — Seismically induced failure of the wall separating the EFW room from the diesel generator rooms (0.89 g) or of the CCW heat exchanger building (1.1 g), when combined with seismically induced failure of the gas turbine, will lead to core damage. If these structures were built with an HCLPF lower than 0.5 g, the plant-level HCLPF would also be lower than 0.5 g (with a lower limit of 0.36 g, the HCLPF value of the CTG).

- Class 1E EDS — Seismically induced failure of a number of important safety-related electrical components, when combined with seismic failure of the gas turbine, would lead to core damage. These include the 125-V dc Class 1E batteries and racks (1.33 g), the EDG supply breakers (0.95 g), the 480-V ESF load centers (0.95 g), and the 480-V load center ESF transformers (0.95 g). If any of these components were installed with an HCLPF lower than 0.5 g, the plant HCLPF would also be lower than 0.5 g (with a lower limit of 0.36 g).

- Component cooling water and SSWSs — Seismic failure of the CCW or SSW pumps, due to the failure of their associated circuit breakers (0.95 g), when combined with seismic failure of the CTG, will lead to core damage. If any of these components were installed with an HCLPF lower than 0.5 g, the plant-level HCLPF would also be lower than 0.5 g (with a lower limit of 0.36 g).

All other seismic sequences require multiple failures of SSCs whose HCLPF is greater than 0.5 g in order to drive the plant to core damage. The capacity of as-built SSCs to meet the HCLPFs assumed in the System 80+ PRA (see CESSAR-DC Section 19.7.5) will be verified by a seismic walkdown. The COL applicant will develop details of that walkdown. This is COL Action Item 19-8.

Cutsets With Both Seismic and Non-Seismic Failures

A review was also conducted to identify seismic/random combinations in which the HCLPF of the seismic portion was less than the seismic-only plant-level HCLPF and in which the random portion was not screened out (i.e., it was $> 1 \times 10^{-3}$). These cases are considered important because small increases in the random failure probability could impact the perception of the plant HCLPF. Put simply, as the probability of random failure approaches 1.0, the plant HCLPF would approach the HCLPF of the seismic portion. Only one such combination was identified.

- Emergency ac power system — Random failure of the EDGs, due to various combinations of failure to start, failure to run, and maintenance unavailability (summing to about 1×10^{-2}), when combined with seismically induced failure of the CTG (0.36 g), will lead to core damage (SBO). As the probability approaches 1.0, the plant-level HCLPF will approach 0.36 g. The staff finds this combination of seismic and random failures as estimated by ABB-CE to be acceptable due to the EDG's high reliability. However, if the random failure rate for the EDGs were to become much higher, it would call into question whether a vulnerability existed in the design. Similarly, if the seismic HCLPF for the CTG were to become significantly lower, the staff would wish to examine if a vulnerability existed. Because of these insights, it is important that the CTG procured by the COL applicant be at least as seismically robust as assumed in the PRA (COL Action Item 19-9) and that the EDGs be appropriately included in the RAP (COL Action Item 19-14). In addition, the structure that houses the CTG must have an HCLPF of at least that of the CTG or must be designed in such a manner that failure of the structure following a seismic event up to the HCLPF of the CTG will not effect the operability of the CTG. This should also be true for any other structure whose failure could affect the availability of the CTG to perform its intended function.

All other seismic/random combinations require either failure of structures or components whose HCLPFs exceed 0.5 g, or random failures whose random failure probability is less than 1×10^{-3}.

Important Human Actions and Random Failures

The same human error rates and random failure rates that were used in the System 80+ internal events analysis were also used in the SMA. The PRA-based SMA did not identify any human reliability insights that were not already identified in the internal events analyses. There were no human actions that contributed to the plant HCLPF value. Only one broad class of random failures contributed to the plant HCLPF value, namely, those that are related to EDG reliability, including failure to start, failure to run, and maintenance unavailability.

19.1.4.1.3 Insights from Uncertainty, Importance, and Sensitivity Analyses (Seismic)

One of the reasons for performing an uncertainty analysis is to display the range of values within which the results of an analysis could reasonably be expected to fall. The use of a PRA-based SMA inherently makes use of the breadth of information being considered. This is because HCLPF values can be thought of as the g-level at which one has 95 percent confidence that less that 5 percent of the time the equipment will fail (i.e., we are dealing with the tails of the curves). It was not found necessary to combine (use convolution) a seismic hazards analysis with equipment fragilities, since hazard curves have a large uncertainty which reduces their value in helping to make judgments about the seismic risk. From seismic PRA analyses, it is clear that uncertainties in the hazard curves would dominate the uncertainties in equipment/structure fragilities. For the System 80+ PRA-based SMA, no uncertainty analysis was performed because uncertainty is directly reflected in the margins method. Also, since the margins method does not result in either CDF or risk results, importance analyses were not performed. ABB-CE did, however, perform sensitivity analyses to evaluate the effects of changes in certain assumptions used in the SMA. These sensitivity analyses covered two areas. The first addressed the sensitivity to changes in the HCLPF values of certain key SSCs. The second addressed the sensitivity to changing the site conditions from rock to various soil types.

The analysis of sensitivity to changes in HCLPF values considered changes in the "generic" HCLPFs (i.e., HCLPF values of SSCs based on generic data) used for three major component classes: motor operated valves, piping, and breakers. It also considered the sensitivity to changes in the design-specific HCLPFs values for four SSCs: the reactor protection system (RPS), the wall separating the EFW room from the EDG rooms, and the CST. Sensitivity to both increases and decreases in the HCLPF values was assessed.

When the HCLPF values of these SSCs are increased, the plant HCLPF value does not change (the plant HCLPF value is controlled by the structural failure of the containment which is 0.73 g). The HCLPF value of the containment structure controls the plant-level HCLPF value up to a value of 0.86 g, when reactor vessel rupture becomes the dominant sequence. In this instance, the increase in the seismic-only plant HCLPF value to such a high level without a similar increase in the dominant seismic/random sequence would result in a determination that the plant HCLPF value is dominated by seismic/random combinations. The dominant sequence for this case would be SBO from the seismic/random combination of [0.36 g (CTG)] [1×10^{-2} (EDGs)].

When the HCLPF values of these SSCs are decreased, most of the SSCs analyzed have an impact on the plant level HCLPF. First, decreasing any of the three "generic" HCLPF values will impact the results. This is not surprising since they affect large numbers of components. There are always one or more sequences whose HCLPF value is controlled by one or more of the components with "generic" HCLPF values, so it is necessary to ensure that these HCLPF values are not inappropriately low in the as-built plant. This will be confirmed by the COL applicant during a seismic walkdown of the as-built plant (COL Action Item 19-8). In addition, a decrease in the HCLPF value of the EFW/EDG room wall causes the plant-level HCLPF value to decrease (as discussed in Section 19.1.4.1.2, the lower limit for this sensitivity is 0.36 g).

A reduction in the HCLPF value of the CST or the RPS, has no effect on the plant-level HCLPF. However, the staff notes that if the as-built CST HCLPF were actually significantly below 0.6 g, it would profoundly increase the number of cutsets where it would take the reduction of only one SSC HCLPF value to reduce the plant HCLPF value below 0.73 g. For this reason, the CST HCLPF in the as-built plant is particularly important.

Sensitivity analyses were also performed for the effects of siting the plant on one of a variety of soil sites (versus the base case of a rock site). Five different soil types were considered. For all cases, the HCLPF values for the dominant sequences and for the plant were unchanged. Some effects were observed for high HCLPF sequences (above 0.85 g), but the staff considers none of these effects significant.

19.1.4.2 Internal Fires Risk Analysis

Because detailed design information was lacking regarding cable routing, fire detection, and the location of suppression systems, ABB-CE chose not to perform a detailed PRA to assess the risk from internal fires associated with the System 80+ design. A detailed PRA involves modeling the propagation of fire, smoke, and hot gases between all the various fire areas containing safety-related equipment. Instead, ABB-CE performed a "scoping" quantitative risk analysis and used it, in conjunction with a qualitative fire analysis, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. This "scoping" analysis considers fires in the nuclear annex and the station SSW/CCW building. Fires in the MCR or in the containment were examined separately using both qualitative and quantitative arguments.

The "scoping" fire risk analysis was based on two key assumptions: (1) integrity of the divisional separation between redundant safety-related equipment (this divisional separation, which is extended in addition to the nuclear annex in the SSWS/CCWS building, prevents fires from propagating from one division to the other); and (2) the conservative assumption that a fire which initiates a transient, causes all safety-related equipment in the division in which the fire occurred to fail. Using these assumptions, ABB-CE estimated a CDF from internal fires that occur when the plant is operating at power to be about 6×10^{-8}/year. This CDF estimate is quite low compared to internal fire CDF estimates for operating plants. This estimate is viewed as an upper bound by ABB-CE. The staff believes that such a conclusion is not possible without a detailed PRA. When CDFs so small are estimated in a PRA, it is the areas of the PRA in which modeling is least complete and supporting data is sparse or even non-existent that could actually be the more important contributors to risk. Areas not modeled or incompletely modeled in virtually every PRA include errors of commission, sabotage, rare initiating events, construction errors, design errors, control systems, aging, systems interactions, and human errors. However, due to the incorporation of design features such as physical separation between safety divisions, the staff believes that the System 80+ design capability to prevent and mitigate severe accident fires is superior to operating plants, and that the conclusions from the high-level "scoping" fire risk analysis performed by ABB-CE complements this belief. The fire risk analysis has provided useful safety insights for inclusion in ITAACs, COL action items, and RAP. Since detailed PRA-based internal fire analyses at some operating plants have shown that fire-induced sequences can be leading contributors to CDF, ABB-CE has stated (COL Action Items 19-5 and 19-12) that the COL applicant should prepare an updated internal fire PRA that takes into account design details, such as cable routing and door locations as well as fire detection and suppression system location, to search for internal fire vulnerabilities in the detailed design.

The staff's review determined that there are 11 doors above the 70+ ft level in the System 80+ design where the door penetrates the dividing divisional wall (i.e., hard concrete interdivisional barrier) between the two divisions. ABB-CE submitted additional details on the System 80+ design to justify that these 11 doors do not constitute potential fire vulnerabilities. ABB-CE stated that all of these doors are self-closing and are alarmed in the MCR. In addition, there are always additional intervening fire doors (self-closing doors and alarms) between the doors in the divisional wall and any safety-related equipment or equipment credited in the System 80+ PRA.

RCP seal LOCAs are not modeled as a credible event in the internal events analysis. Instead ABB-CE prepared a sensitivity study for at-power events on the potential effects of loss of cooling to the seals. However, because the staff was concerned about the potentially much higher common-mode failure rate associated with fire events, ABB-CE evaluated loss of seal cooling due to a fire and resultant RCP seal LOCA. On the basis of the results reported in this submittal (very low probability of core damage), the staff finds that RCP seal LOCAs do not constitute a vulnerability during fires that result from severe accidents. However, the staff believes that the available test results for the RCP seals do not provide full confidence in the capabilities of the seals under loss-of-seal-cooling conditions. Therefore, for the purposes of defense in depth, the dedicated seal injection pump (an air-cooled positive displacement pump) should be located in such a manner as to minimize its vulnerability to internal fires and floods that could also affect the primary means of providing RCP seal cooling or RCP seal injection (see CESSAR-DC Table 19.15-1, "Significant PRA-based safety insights for System 80+").

The System 80+ design has significant robustness to prevent and mitigate severe-accident fires and the design should result in a plant with superior capabilities to prevent and mitigate fires compared to operating nuclear power plants.

19.1.4.2.1 Dominant Accident Sequences Leading to Core Damage (Internal Fires)

ABB-CE's "scoping" fire-risk analysis for the System 80+ design used applicable event and fault tree models from the internal events analysis (equipment needed to mitigate "transient" events) to identify the following dominant accident sequences:

Sequence 1, the fire disables one division of ESFs; there is a failure of the DHR system; and there is failure of SDS valves to bleed so feed-and-bleed cannot be performed.

Sequence 2, the fire disables one division of ESFs; there is a failure of the DHR system; and the SI, which is needed for injection, fails.

Sequence 3, the fire disables one division of ESFs; there is a failure of all the feedwater systems (main and auxiliary); and there is a failure of SDS valves to bleed so that feed-and-bleed cannot be performed.

Sequence 4, The fire disables one division of ESFs; there is a failure of all the feedwater systems (main and auxiliary); and the SIS fails to provide injection.

19.1.4.2.2 Risk Important Design Features and Requirements (Internal Fires)

The System 80+ design incorporates many features for detecting and confining fires. When considering the diagrams of the nuclear island in the CESSAR-DC, it is quite apparent at a rudimentary level that the System 80+ design has design features that provide significant mitigation capability to prevent fires or their byproducts from spreading. This is because the System 80+ design was developed with the expectation that fires would need to be detected, suppressed, and prevented from spreading. The most important of these design features is the reinforced-concrete wall between the two safety divisions that provides a floor to ceiling barrier rated for at least a 3-hour fire up to elevation 70+ ft. The only places where this wall is breached (up to elevation 70+ ft) is where several CVCS pipes penetrate. These penetrations must have the same fire rating as the concrete wall. Exceptions to physical separation between divisional equipment are in the MCR, in a stairway between the MCR and the remote shutdown panel, at the remote shutdown panel, and in the containment. Availing itself of engineering judgment, the staff has identified the following major design features as responsible for reducing the likelihood of a fire leading to core damage in the System 80+ design:

- Physical separation between the two divisions of equipment in the nuclear annex is very important. The wall is rated as a 3-hour-rated fire barrier. There are penetrations but no such openings in the divisional wall, as doors or heating, ventilation, and air conditioning (HVAC) ducts, at the very lowest level of the plant where most of the important safety equipment resides. However, higher up in the plant (elevation level 70+ ft and above) there are 11 doors through which a fire could propagate.

- On either side of each of the 11 doors that penetrate the dividing divisional wall are additional fire doors. They are located so that any fire would need to pass through at least three fire doors to affect equipment in both divisions that is safety-related or credited in the PRA.

- There are separate ventilation systems for the MCR and the remainder of the control building areas, including the location of the remote shutdown panel and the Divisions I and II electrical systems.

- There are separate ventilation systems for the Divisions I and II systems within the nuclear annex.

- Air intakes for the ventilation systems are located so as to minimize the exposure to smoke, hot gases, and fire suppressants resulting from a fire in other areas. Separate systems are provided for each division.

- In the event of a fire requiring the evacuation of the MCR, all equipment used in response to a transient initiator can be operated either directly at the remote shutdown panel or via the operator module at the remote shutdown panel. Transfer of control to the remote shutdown panel is accomplished by controls located near both MCR exits.

- The cable spreading room has been eliminated from the System 80+ design.

- No safety-related equipment is located in the turbine building and there is a 3-hour-rated fire barrier between the turbine building and the nuclear annex.

- Redundant transformer yards are physically separated so that they are not susceptible to CCF from a single fire.

- Materials used in the MCR control panels are of materials that would not independently support combustion; neoprene cannot be used in the panels; and PVC use is limited. However, there are materials stored in the MCR that will support a fire, such as procedures and other paper documents.

- There are 3-hour-rated fire barriers between quadrants, and the power and control cables to the quadrants are separated by 3-hour-rated fire barriers.

- All 3-hour-rated fire barriers are seismic Category I and are made of reinforced concrete with doors that automatically close and are alarmed to the MCR.

- There are no HVAC dampers in barriers that separate redundant divisions of safety-related equipment.

- HVAC chases contain only HVAC ductwork and dampers. No instrumentation, cables, valves, or other components are located in the HVAC chases.

The human actions modeled in the internal fire analysis are the same as those modeled in the internal events analysis.

No additional activities were identified that would result in significant human reliability insights.

Because the approach taken in performing the System 80+ internal fire analysis makes various conservative assumptions, performance of an uncertainty, sensitivity, or importance analyses would result in biased insights. Since the purpose of performing an uncertainty analysis is to better understand the subject being investigated and since, in the case of the internal fire analysis, it is unclear what the results of an uncertainty analysis would represent physically given the nature of the assumptions in the analysis, an uncertainty analysis is not appropriate.

19.1.4.3 Internal Flooding Risk Analysis

Due to lack of detailed design information needed to identify the potential flood sources and flood levels, such as pipe routing, flood curbs, and flood barriers, ABB-CE chose not to perform a detailed PRA to assess the risk from internal flooding associated with the System 80+ design. A detailed PRA involves modeling the propagation of flooding, between all the various flood areas containing safety-related equipment. Instead, ABB-CE performed a "scoping" quantitative risk analysis and used it, in conjunction with a qualitative flood analysis, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. This "scoping" analysis considers floods in the nuclear annex and the SSWS/CCWS building. The potential for flood propagation from the turbine building to the nuclear annex, where safety-related equipment is located, was also examined using both qualitative and quantitative arguments.

The "scoping" flood risk analysis was based on two key assumptions: (1) integrity of the divisional separation between redundant safety-related equipment (this divisional separation, which is extended in addition to the nuclear annex in the SSWS/CCWS building, prevents floods from propagating from one division to the other); and (2) the conservative assumption that a flood which initiates a transient, causes all safety-related equipment in the division where the flood occurred to fail. Using these assumptions, ABB-CE estimated a CDF from internal floods that occur when the plant is operating at power to be about 1×10^{-8}/year. This CDF estimate is quite low compared to internal flood CDF estimates for operating plants and ABB-CE views this as an upper bound. The staff believes that such a conclusion is not possible without a detailed PRA. The staff did not concentrate its review on bottom-line numbers but rather on the relative insights that the internal flood analysis provides. The staff believes that the System 80+ design is capable of preventing and mitigating severe-accident internal floods in a manner superior to operating plants and that the conclusions from the "scoping" internal flood risk analysis performed by ABB-CE complement this belief. The internal flood risk analysis has provided useful safety insights for inclusion in ITAACs, COL action items, and RAP. Since detailed PRA-based internal flood analyses at some operating plants have shown that flood-induced sequences can be leading contributors to CDF, ABB-CE has stated that the COL applicant (COL Action Items 19-12 and 19-5) should provide an updated internal flood PRA that takes into account design details, such as pipe routing, door locations, and flood barriers, to search for internal flooding vulnerabilities in the detailed design.

The most important features of the System 80+ design with respect to preventing and mitigating internal floods are (1) the use of only closed cooling water systems in the nuclear annex (there is no cross-connect between the divisions of CCW), (2) the reinforced-concrete divisional wall between the divisions in the nuclear annex that has no doors or passages below elevation 70+ ft, and (3) flood barriers between the quadrants in the subsphere to help limit internal floods to one quadrant.

RCP seal LOCAs are not modeled as a credible event in the internal events analysis. Instead ABB-CE prepared a sensitivity study for at-power events on the potential effects of loss of cooling to the seals. However, due to the staff's concern about the potentially much higher common-mode failure rate associated with flood events, ABB-CE evaluated loss of seal cooling due to a flood and resultant RCP seal LOCA. On the basis of the results reported in this submittal (very low probability of core damage), the staff finds that RCP seal LOCAs do not constitute a vulnerability during severe-accident floods. However, the staff believes that the available test results for the RCP seals do not provide full confidence in the capabilities of the seals under loss-of-seal-cooling conditions. Therefore, for the purposes of defense in depth, the diverse positive displacement RCP seal injection pump (air-cooled) should be located in such a manner as to minimize its vulnerability to internal fires and floods that could affect that could also affect the primary means of providing RCP seal cooling or RCP seal injection (see CESSAR-DC Table 19.15-1, "Significant PRA-based safety insights for System 80+").

The System 80+ design has significant robustness to prevent and mitigate severe-accident floods and the design should result in a plant with superior capabilities to prevent and mitigate floods compared to operating nuclear power plants.

19.1.4.3.1 Dominant Accident Sequences (Internal Floods)

ABB-CE's "scoping" flood-risk analysis for the System 80+ design used applicable event and fault tree models from the internal events analysis (equipment needed to mitigate "transient" events) to identify the following dominant accident sequences:

- Flood disables one division of ESF, there is failure of the DHR system, and failure of SDS valves to reduce pressure to perform feed-and-bleed.
- Flood disables one division of ESF, there is failure of the DHR system, and the SI pump fails to provide feed for feed-and-bleed.
- Flood disables one division of ESF, there is failure of all feedwater systems, and failure of safety depressurization valves to reduce pressure so can perform feed-and-bleed.
- Flood disables one division of ESF, there is failure of all feedwater systems, and the SI fails to feed for feed-and-bleed.

19.1.4.3.2 Risk Important Design Features and Human Actions (Internal Floods)

The following is a list of some of the design features that are responsible for reducing the impact of an internal flood in the System 80+:

- There are no sources of "unlimited" quantity of water within the nuclear annex. The connections to the "unlimited" sources of water in the System 80+ design are external to the nuclear annex. There are no paths through which the water from these external, "unlimited" sources can enter the nuclear annex or RB. The SSWS is an important "unlimited" source of water, and it interfaces with the CCWS in the SSWS/CCWS heat exchanger structure. Similarly, unlimited sources of water in the turbine building cannot enter the nuclear annex.
- Divisional separation is maintained within the nuclear annex, outside containment. There are no doors or passageways connecting the divisions up to elevation 70+ ft. Any penetrations (e.g., CVCS piping) through the divisional wall has a watertight seal.
- Entrances from the turbine building to the nuclear annex are located above the maximum turbine building flood level (39 ft above the turbine building grade level).
- Drains in the nuclear annex and the RB are divisionally separated, have seismic Category I valves to prevent backflow, and are sized to handle fire-protection system discharges. Each subsphere quadrant has its own redundant seismic Category I sump pumps that can be powered off the EDGs.
- There are no cross-connections in closed-loop cooling water systems. This limits the effects of floods to the operability of systems in one division.
- Flood sources in the nuclear annex, other than the fire-protection system, are all located below elevation 70+ ft.
- The turbine building contains no safety-related equipment.
- No water lines are routed above or through the MCR and the computer room. HVAC water lines contained in rooms around the MCR are located in rooms with raised curbs to prevent leakage from entering the MCR.
- There are flood barriers between quadrants, and the power and control cables to the quadrants are separated by flood barriers.
- The CCW heat exchanger building and the station service water structure have reinforced concrete separating divisional walls to protect against interdivisional floods. The COL applicant is to provide a site-specific PRA to evaluate the design of these buildings to determine if there are any vulnerabilities to internal floods or fires. This is COL Action Item 19-12.
- The COL applicant should provide an updated internal flood PRA that takes into account design details, such as pipe routing, door locations, and flood barriers, to search for internal flooding vulnerabilities in the detailed design. This is COL Action Item 19-12.
- The COL applicant should provide a site-specific PRA-based external flood analysis (COL Action Item 19-12) (SECY-93-087) to determine if the siting of the plant has introduced any vulnerabilities due to severe-accident external floods.

The human actions modeled in the internal flood analysis are the same as those modeled in the internal events analysis. No additional activities were identified that would result in significant human reliability insights.

Because the approach taken in performing ABB-CE System 80+ internal flood analysis makes various conservative assumptions, performance of an uncertainty, sensitivity, or importance analyses would result in biased insights. Since the purpose of performing an uncertainty analysis is to better under-stand the subject being investigated and since, in the case of the internal flood analysis, it is unclear what the results of an uncertainty analysis would represent physically, given the nature of the assumptions in the analysis, an uncertainty analysis is not appropriate.

19.1.4.4 Tornado-Risk Analysis

ABB-CE estimates the CDF due to the "tornado" category of external events (which includes also high winds and hurricanes, in addition to tornado-generated missiles) was to be about 3×10^{-7}/year. It was assumed that a tornado strike event at the site would result in (1) the LOOP with a duration in excess of 24 hours, (2) the loss of the turbine-generator runback capability to pick up hotel loads, and (3) the loss of the non-safety-grade CTG. These assumptions imply that, following the tornado strike, the EDGs are required to operate for at least 24 hours. The staff believes this analysis is adequate for the purpose of identifying design vulnerabilities from tornado events as well as identifying certification requirements and COL action items. Section 19.1.4.4.1 of this chapter presents the dominant accident sequences and the major contributors to CDF estimates from tornado strike events; Section 19.1.4.4.2 presents those System 80+ design features which reduce or eliminate vulnerabilities to tornado events associated with operating nuclear power plants.

19.1.4.4.1 Dominant Accident Sequences Leading to Core Damage (Initiated by Tornado Events)

Two dominant accident sequences contribute ~99 percent of the CDF estimate from tornado strike events. Sequence 1, with a CDF estimate of 2.4×10^{-7}/year (a 94-percent contribution), is initiated by a tornado strike event which causes LOOP without possibility for recovery within 24 hours as well as loss of the non-safety-grade AAC power source (CTG). Failure to remove decay heat using either the EFWS or the SCS (once shutdown cooling (SDC) entry conditions are met) requires feed-and-bleed operation. Failure of the "feed" portion of the feed-and-bleed operation (i.e., failure of SIS) leads to core damage. This sequence is dominated by operating failures (failure to run) of both EDGs. The dominant causes of failure of both EDGs are (1) blockage of the service water intake due to tornado-generated debris (causes loss of cooling water to the EDGs and eventually their failure to run), (2) CCF of the EDG to run for at least 24 hours, and (3) independent failure of both EDGs to run for at least 24 hours.

Sequence 2, with a CDF estimate of 1.4×10^{-8}/year (a 5-percent contribution), is initiated by a tornado strike event which causes a LOOP (which cannot be recovered during 24 hours) as well as loss of the CTG. Failure of both EDGs to start leads to the depletion of station batteries in about 8 hours and to core damage in about 10 hours. The major contributors to this sequence are common-cause and independent failure of both EDGs to start on demand.

19.1.4.4.3 Risk-Important Design Features and Requirements (Tornado Events)

The following are the most important features contributing to the lower CDF, associated with tornado strike events, of the System 80+ design as compared to operating nuclear power plants:

- There is no safety-related equipment in the turbine building (consequential failures of safety equipment located in the turbine building and LOOP are major contributors to risk in currently operating plants).

- The use of reinforced concrete for the station service water (SSW) intake structure reduces tornado missiles, primarily those associated with masonry block construction. This reduces or eliminates such typical failures (associated with tornado-induced missile strikes on the SSW intake structure in operating plants) as failures of SSW pumps and suction piping, and structural collapse of the intake structure blocking intake flow.

- The System 80+ design includes two EFWSTs which are located in the nuclear annex (a reinforced-concrete structure) and below grade level. This eliminates a tornado-induced failure of the EFWSTs, the System 80+ primary source of EFW for DHR. In operating plants, this failure is a possibility since current-generation plants have their primary source of EFW for DHR (usually a CST) located in or near the turbine building.

- All transformers for the safety-related equipment are located within reinforced-concrete structures (most operating plants have the nonessential and step-down transformers located in or just outside the turbine building which provides no protection from tornados.

- The System 80+ design has a secondary switchyard connected to the grid at a remote location (no credit was taken for this feature in the tornado PRA).

An important safety insight gained from the tornado analysis is that "blockage of the service water intake flow by tornado-generated debris" is a major contributor to risk from tornados. The COL applicant should evaluate the vulnerability of the SSW intake to tornado-generated debris. This is COL Action Item 19-1).

19.1.5 Safety Insights From the Risk Analysis for the Low-Power and Shutdown Operation

ABB-CE assessed the risk associated with LP&S modes of operation for internal events as well as for internal fires and floods. ABB-CE used methodologies, results, and insights from available recent studies performed for operating PWRs. Although there are limitations in the "state of the art" associated with probabilistic risk assessment methodologies and data for LP&S operation, the major objectives have been addressed. These objectives were (1) identify design and operational vulnerabilities related to LP&S operation and (2) identify risk-important design features, plant configurations, human actions, and operational requirements.

The LP&S risk assessment for the System 80+ design covers operational Mode 4 (hot shutdown), Mode 5 (cold shutdown), and Mode 6 (refueling). The risk associated with Mode 2 (reactor critical at less than 5 percent power) and Mode 3 (hot standby) was not assessed. This is justified because the plant configurations in Modes 2 and 3 are very similar to Mode 1 (power operation) and the fraction of time the plant is in Modes 2 and 3 is negligible compared to the fraction of time the plant is in Mode 1. Because of the various plant configurations during shutdown operation, it was necessary to define different plant operational states (POSs) based on plant response and the equipment that is available to mitigate the event. Some POSs include more than one plant operation mode. The following four POSs were defined:

- POS 1: The plant is in Mode 4 or in Mode 5 with normal inventory or in Mode 6 with the IRWST full and refueling cavity empty.
- POS 2: The plant is in Mode 5 with reduced inventory (5R), including "midloop" operation where the coolant has been reduced to the midpoint of the hot leg so that SG nozzle dams can be installed.
- POS 3: The plant is in Mode 6 with the IRWST empty, the refueling cavity full, and the upper internals removed (6E).
- POS 4: The plant is in Mode 6 with IRWST empty, refueling cavity full, and the upper internals in place (6I).

The contributions to the estimated CDF for LP&S operation, by POS and initiating event category, are summarized in Table 19.4. The leading contributor from internal events is POS 2, that is, the plant is in Mode 5 with reduced inventory (~48 percent contribution). The plant configuration in Mode 6 with the IRWST empty (POSs 3 and 4) is the next dominant contributor (~30 percent contribution to the CDF estimate from internal events during shutdown). The third leading contributor is POS 1 (primarily due to ~20 percent contribution), primarily due to the increased likelihood of a LOOP because of the long time spent in plant configurations included in this POS. With respect to initiating events, LOOP is the leading contributor to the estimated CDF from internal events during shutdown (~39 percent), followed by loss of DHR (~36 percent) and LOCA (~25 percent).

19.1.5.1 Dominant Accident Sequences Leading to Core Damage at LP&S Operation

The following four dominant accident sequences were identified in the shutdown risk analysis (they contribute approximately 75 percent of the total CDF from internal events during shutdown).

Sequence 1, with a CDF of 1.3×10^{-7}/year (a 24-percent contribution), is initiated by the loss of DHR when the plant is in Mode 5R (reduced reactor coolant inventory). Failure to restore the operating SCS train and to use either a charging pump or an SCS pump to makeup coolant inventory requires the establishment of a feed-and-bleed operation for core cooling. Failure of the SIS to provide the "feed" portion of the feed-and-bleed operation leads to core damage.

Sequence 2, with a CDF of 1.0×10^{-7}/year (a 19-percent contribution), is initiated by a LOOP event during Mode 4 and Mode 5 with normal coolant inventory and Mode 6 with IRWST full and refueling cavity empty. If all three onsite ac power sources (i.e., two EDGs and a CTG) are unavailable, failure to restore ac power within certain timeframes leads to core damage.

Sequence 3, with a CDF of 8.6×10^{-8}/year (a 16-percent contribution), is initiated by a LOCA inside the containment during refueling with the IRWST empty and the refueling cavity full. The operator fails to isolate the leak and must establish a "feed-and-bleed" operation to cool the core. Since the coolant from the leak drains into the IRWST, it is available for injection or "feed" using any of the SCS or CSS pumps. Failure to provide injection using an SCS or CSS pump, requires use of a CVCS charging pump for the "feed" operation to at least match core boil-off and maintain core cooling. In this sequence the "feed" operation from the boric acid storage tank (BAST) using a CVCS charging pump is successful. However, if the operators fail to restore an SCS train before the inventory in the BAST is depleted, core damage may ensure.

Sequence 4, with a CDF of 8.2×10^{-8}/year (a 15-percent contribution), is initiated by a LOOP during plant operation in Mode 5 with reduced inventory. If all three onsite ac power sources (i.e., two EDGs and a CTG) are unavailable, failure to restore ac power (within a certain timeframe) leads to core damage.

Table 19.4 CDF from Internal Events During Shutdown

POS	CDF by Initiating Event			Total CDF (per year)
	Loss of DHR	LOCA	LOOP	
#1 (4, 5, 6F)	1×10^{-9}	1×10^{-9}	1×10^{-7}	1×10^{-7}
#2 (5R)	2×10^{-7}	3×10^{-8}	8×10^{-8}	3×10^{-7}
#3 (6E)	7×10^{-9}	7×10^{-8}	7×10^{-9}	8×10^{-8}
#4 (6I)	3×10^{-8}	4×10^{-8}	3×10^{-9}	7×10^{-8}
Total	2×10^{-7}	1×10^{-7}	2×10^{-7}	5×10^{-7}

19.1.5.2 Risk-Important Design Features and Operational Requirements at LP&S Operation

The following are important factors contributing to the decrease or elimination of vulnerabilities in the System 80+ design, during shutdown operation, as compared to current-generation plants.

- Defense in depth which provides alternative means to the operator for maintaining coolant inventory and removing decay heat during a LOCA or a loss of SDC (loss of DHR) event. Important design features and operational requirements are:
  - Two separate and independent divisions of the SCS. In addition, the SCS and the CSS pumps are independent but identical and functionally interchangeable. This characteristic contributes to the increased availability of these two systems, as compared to operating reactor designs, to maintain coolant inventory and/or remove decay heat.
  - The SCS can be aligned to the IRWST during plant shutdown operation to provide inventory makeup or to perform a feed-and-bleed operation.
  - If all SCS/CSS pumps are unavailable for DHR and/or coolant inventory makeup, the operator can still perform these functions by feed-and-bleed using the SDS or the LTOP valves for the "bleed" function and the SIS or the CVCS charging pumps for the "feed" function.
  - SI for feed-and-bleed is an important DHR alternative during shutdown operation. As a result of this, a new TS was added requiring that two of the four SIS pumps be available in shutdown modes when the IRWST is available.
- Design features which are important for preventing and mitigating LOOP/SBO events during power operation are also important in reducing the frequency of these events during shutdown operation. These are (1) two separate and independent switchyards and (2) redundant and diverse onsite AAC power sources (two EDGs and

a CTG). The following operational requirements are important during shutdown operation:

- The reliability of the two switchyards is an important feature contributing to the reduced System 80+ shutdown risk from LOOP/SBO events, as compared to operating reactor designs. When a switchyard is unavailable for maintenance, the COL applicant should ensure that no activities which could cause the operating switchyard to fail are taking place and no transient fire sources are present. This is COL Action Item 19-17.

- The reliability of the redundant and diverse emergency onsite ac power sources is an important feature contributing to the reduced shutdown risk from LOOP/SBO events, as compared to operating reactor designs. For this reason, a new TS was added requiring that two of the three onsite emergency ac power sources (i.e., two EDGs and a CTG) be available during shutdown operation.

- Because plant shutdown places increased reliance on human actions and creates greater opportunity for human errors, risk can be minimized by appropriate outage management, administrative controls, procedures, training, and operator knowledge of plant configuration. The control of these activities is an important COL applicant responsibility. This is COL Action Items 19-15 and 19-17.

- During plant shutdown, the integrity of fire and flood barriers should be maintained between areas in the same division, such as quadrants, where systems comprising the alternate shutdown success paths are located. This will require configuration control of fire/flood barriers for shutdown operation by the COL applicant. The COL applicant should incorporate in its configuration control program a requirement that at least one quadrant within the subsphere (with at least one SCS/CSS pump operable) be maintained physically isolated with its water-tight fire doors closed during Modes 4, 5, and 6 to help prevent common-mode failures from internal floods or fires. This is COL Action Item 19-18.

19.1.5.3 Insights From the Importance and Sensitivity Analyses for LP&S Operation

ABB-CE performed an importance analysis for LP&S operation with similar objectives as for power operation, that is, risk reduction and safety or reliability assurance. Because of less detailed models of system failures and interactions during shutdown than at power operation, ABB-CE performed the importance analysis at the system or function level rather than at the component or event level. Nevertheless, important insights were drawn from this analysis. In addition, ABB-CE conducted selected sensitivity analyses to provide insights about the impact of uncertainties on the estimated CDF during shutdown.

<u>Insights From the Importance Analysis</u>

The importance analysis for LP&S operation addresses two objectives: (1) risk reduction and (2) safety or reliability assurance. The first objective, that is, insights on risk reduction, was achieved by the identification and ranking of system functions, human actions, and accident-initiating events that are dominant contributors to the estimated risk during shutdown (having highest "risk reduction worth"). The second objective, that is, safety or reliability assurance, was achieved by identifying and ranking system functions and human actions required during shutdown which contribute the most in maintaining the "designed-in" risk level (having highest "risk achievement worth"). Details on SSCs and human actions that ABB-CE found risk significant are documented in CESSAR-DC Section 19.15. This information, which was generated by taking into account insights and assumptions from the entire PRA (i.e., all three PRA levels for both internal and external events and for all modes of operation), form the basis for the following two lists: (1) a list of important SSCs (CESSAR-DC Section 19.15.6.1) which the COL applicant should incorporate in the D-RAP and O-RAP (COL Action Item 19-14), and (2) a list of risk-important ("critical") operator tasks (CESSAR-DC Table 19.15.6-2) which should be taken into account in the MCR design as well as in developing detailed procedures and training programs. These lists were incorporated into the CESSAR-DC in Amendment V and are acceptable. This portion of FSER Confirmatory Item 1.1-1 is resolved.

The following system functions and human actions were found to have high "risk achievement worth." Note that most of these system functions require operator action:

- Isolate a leak outside containment (ISLOCAs).
- Align the SCS to the IRWST to provide inventory makeup or to perform a feed-and-bleed operation when all SIS pumps are unavailable.
- Start and run an EDG following a LOOP event.
- Load and start the standby CTG if both EDGs are unavailable following a LOOP event.
- Isolate a leak locally (many LOCAs during shutdown are caused by operator errors in aligning valves and, therefore, the leak can be isolated locally).

- Start the standby SCS train when the operating train fails and the operator cannot recover it in a timely manner.

- Use an SIS pump to provide the "feed" portion of the feed-and-bleed operation during a loss-of-DHR accident in Mode 5R (reduced inventory, including "midloop" operation).

- Recover the operating SCS train soon after it fails (many of the causes for loss of DHR are operator errors that can be recovered).

- Restore an SCS train in order to recover DHR, following successful "feed" operation using a CVCS charging pump, before the BAST inventory is depleted (long-term recovery).

- An LTOP valve opens to provide the "bleed" portion of the feed-and-bleed operation when both RDS valves fail to open.

- Provide inventory makeup from the BAST using a CVCS charging pump during a loss-of-DHR event in Mode 5R (reduced inventory). This function is required to increase the reactor coolant inventory before starting the standby SCS train, thus avoiding pump cavitation.

Operational requirements assumed in the PRA (such as maintenance, training, outage management, configuration control, and procedures) should be implemented to ensure that the success probabilities of these high "risk achievement worth" functions and human actions are maintained.

ABB-CE determined that the following system functions and initiating events are the major contributors to the estimated risk for shutdown operation, that is, they have high "risk reduction worth." Note that most of these system functions and initiating events involve human actions.

- Restore an SCS train in order to recover DHR, following successful "feed" operation using a CVCS pump before the BAST inventory is depleted. This recovery is dominated by human actions. The existence of appropriate procedures, training, and spare parts inventory is important.

- Align the SCS to the IRWST to provide inventory makeup or to perform a feed-and-bleed operation when all SIS pumps are unavailable. Major failures associated with this system function are:

  - Operator actions needed to align the SCS for injection and start a pump.

  - Hardware failures (MOVs fail to open)

- Start and run an EDG following a LOOP event.

- Load and start the standby CTG when both EDGs are unavailable during a LOOP event.

- Recover operating SCS train soon after it fails (many of the causes of loss of DHR are operator errors that can be recovered).

- Start the standby SCS train when the operating train fails and the operator cannot recover it. The major contributor to the failure of this function is operator error.

- Loss of DHR initiating event in Mode 5R (with reduced inventory, including "midloop" operation).

- Use an SIS pump to provide inventory makeup or the "feed" portion of the feed-and-bleed operation. This function is particularly important for loss of DHR during operation with reduced inventory (Mode 5R).

- Provide inventory makeup from the BAST using a CVCS charging pump during a loss of DHR accident in Mode 5R (reduced inventory). This function is required to increase the reactor coolant inventory before starting the standby SCS train, thus avoiding pump cavitation.

- LOCA inside containment as the initiating event.

Operational requirements on systems and human actions needed to perform these high "risk reduction worth" functions (such as improved testing and maintenance, improved training, and procedures), which aim at minimizing equipment unavailability and failures as well as the probabilities of human errors, will be the most beneficial in terms of reducing plant risk. Similarly, operational requirements, such as outage management, which aim at minimizing the frequency of accident-initiating events with high "risk reduction worth" will be the most effective action in further reducing plant risk during shutdown operation.

Insights From the Sensitivity Analyses

The most important insights drawn from the sensitivity analyses for shutdown operation, performed by ABB-CE, are summarized below.

- The CDF estimate is sensitive to changes in the frequency of such accident-initiating events during shutdown as LOOP, LOCAs, loss of DHR, and internal fires. This shows the importance of appropriate outage management programs to minimize mistakes which cause these initiating events to happen.

- Reduced inventory (Mode 5R) is the most critical operation during shutdown. Use of nozzle dams for SG maintenance, as a method of limiting the time spent in Mode 5R, has a significant impact on the estimated CDF for shutdown operation. (The total shutdown CDF from internal events would double if nozzle dams are not used.)

- The estimated CDF for shutdown operation is not very sensitive to reasonable changes in the long-term recovery of an SCS train. If the probability of "failure to restore an SCS train, to recover DHR following successful feed using a CVCS pump before the BAST inventory is depleted" is increased by a factor of 5, the estimated shutdown CDF from internal events is increased by a factor of 2.

- The estimated shutdown CDF is not sensitive to assumptions on leak location (i.e., percent of leaks inside versus outside containment) during plant operation in Mode 6 (refueling) with the IRWST empty and the refueling cavity full.

19.1.6 Use of PRA in the Design Process

ABB-CE used PRA in the design process to achieve the following objectives: (1) identify and quantify vulnerabilities in operating reactor designs and introduce features and requirements that reduce or eliminate these vulnerabilities, (2) quantify the effect of new design features and requirements on plant risk in order to confirm the risk reduction credit for these improvements, and (3) select among alternative features or design options.

ABB-CE used PRA insights from both operating reactor experience and the System 80 design (from which the System 80+ design evolved), to identify potential vulnerabilities in operating reactor designs. This information was used to introduce the special evolutionary design features, described in Section 19.1.2 of this chapter, and make the transition from the System 80 to the System 80+ design. Once these features were introduced, PRA was used to quantify their effect on risk and confirm acceptable reduction or elimination of vulnerabilities, including compliance with applicable risk goals. Examples are the CDF reduction estimates (by accident-initiating event category) and associated System 80+ features which contribute to such reduction, reported in Section 19.1.3.1.2 of this chapter.

The following are examples of ways in which ABB-CE modified the System 80+ design or requirements based on the System 80+ PRA and its evaluation by the staff:

- ABB-CE added shear bars to provide a positive connection between the containment shell and the concrete embedment to increase the HCLPF value for containment slipping/overturning. This raised the HCLPF value for this core-damage event from below 0.6 g, which was controlling the plant level HCLPF value, to 0.73 g.

- SI for feed-and-bleed was found by means of the PRA to be an important DHR alternative during shutdown operation. As a result of this, a new TS was added requiring that two of the four SIS pumps be available in shutdown modes when the IRWST is available.

- Through the PRA, ABB-CE determined that the reliability of the redundant and diverse onsite emergency ac power sources is an important feature contributing to the reduced System 80+ shutdown risk from LOOP/SBO events, as compared to operating reactor designs. For this reason, a new TS was added requiring that two of the three onsite ac power sources (i.e., two EDGs and a CTG) be available during shutdown operation.

The following are specific examples of confirmatory use of PRA in the design process:

- ABB-CE used the PRA to improve the reliability of the final design of the reactor CFS. This was achieved by incorporating the following features into the design: (1) the addition of an HVT between the IRWST and the reactor cavity to reduce the potential for inadvertent cavity flooding, (2) the provision of parallel flow paths for transferring water into and out of the HVT (four parallel lines between the IRWST and HVT, and two parallel lines between the HVT and the reactor cavity, each with a single MOV), and (3) use of ac-motor operated valves with dedicated inverters (power from 125-V dc) to provide increased reliability. With these features, the hardware reliability of the CFS was estimated to be about 1×10^{-3}.

- In the original System 80+ PRA, accidents in which the containment failed before core melt were found to be dominant contributors to the frequency of containment failure. As a result of this insight, ABB-CE added the ECSBS. This system provides an independent, self-contained means of supplying water to the containment spray header during an emergency condition when the normal CSS is not available. The system includes an external pumping device taking suction from an external source of water, and delivering the water to the spray headers. Based on the updated System 80+ PRA, the external source of water to the sprays virtually eliminates sequences in which core damage occurs as a result of containment failure.

The following are some specific examples of use of PRA to select among alternate design features or options:

- ABB-CE used PRA to select between two alternative design options for the CCW/station service water (SSW) systems. The first option had standby CCW/SSW systems for cooling safety-related loads. In the second option, the one that was selected, the CCW/SSW systems have two divisions with a normally operating and a standby pump in each division. The selected option eliminates demand failures of pumps and valves in these systems which are known to be significant risk contributors in operating reactors. A subsequent evaluation was also made to determine if the standby pumps had to be automatically loaded onto the EDGs and started following a LOOP event. This evaluation indicated that there would be little risk impact if the standby pumps were aligned to the EDGs following a LOOP but were not started unless the previously running pump failed to restart. Thus, larger and potentially less reliable diesels were not required.

- PRA was also used to select between two emergency ac power design options, namely, "four diesels" versus "two diesels plus a "combustion turbine" configurations. This comparison indicated that the first option was slightly, but not significantly, more reliable than the second option. The second option was selected because it provides power also to the permanent non-safety-related loads and has a much smaller impact on plant size and layout.

Finally, PRA was used to evaluate several severe accident mitigation design alternatives (SAMDAs) by examining the benefits associated with each of these design alternatives.

19.1.7 PRA Input to the Design Certification Process

PRA was used in the design certification process to achieve the following objectives: (1) develop an in-depth understanding of design robustness and tolerance of severe accidents initiated by either internal or external events, (2) develop a good appreciation of the risk significance of human errors associated with the design, and characterize the key errors in preparation for better training and refined procedures, and (3) identify important safety insights and assumptions to support certification requirements, such as ITAACs, design reliability assurance program (D-RAP) and operation reliability assurance process (O-RAP) requirements, TS, as well as COL and interface requirements.

The first two objectives were achieved by identifying the dominant accident sequences as well as the risk-important design features and human actions (see Sections 19.1.3 to 19.1.5 in this chapter). The third objective was achieved by using PRA insights and assumptions to develop the following list of design certification requirements. The specific type of requirement, for example, ITAAC, RAP, and COL is indicated in the text with brackets.

<u>Plantwide requirements</u>

The COL applicant should perform a seismic walkdown to ensure that the as-built plant matches the assumptions in the System 80+ PRA-based SMA and to assure that seismic spatial systems interactions do not exist. The COL applicant will develop details of the seismic walkdown. This is COL Action Item 19-8.

ABB-CE will maintain a list of the SSC HCLPF values used in the System 80+ seismic margins assessment in the D-RAP.

The COL applicant should incorporate the list of important SSCs (see CESSAR-DC Table 19.15.6-1) in its D-RAP and O-RAP. This is COL Action Item 19-14. The D-RAP and O-RAP should reflect the assumptions in the System 80+ PRA regarding SSC testing frequencies and unavailabilities.

The COL holder should have an O-RAP based on the system reliability information derived from the PRA and other sources.

The COL applicant should compare the as-built SSC HCLPF values to those assumed in the System 80+ SMA. Deviations from the HCLPF values (or assumptions) in the SMA should be evaluated by the COL applicant to determine if any vulnerabilities have been introduced. This is COL Action Item 19-9.

The COL applicant should incorporate the information on risk important operator tasks (see list in CESSAR-DC Table 19.15.6-2) in the MCR verification and validation process. The COL applicant should also use this information in developing and implementing procedures, training, and other human reliability related programs. This is COL Action Item 19-15. The COL applicant is also responsible for developing detailed procedures for actuation and operation of the severe-accident design features, such that the use of these features is consistent with the PRA assumptions, or for modifying the site-specific PRA to match the modified assumptions. This is COL Action Item 19-16.

Integrity of divisional separation between redundant safety-related equipment is a key assumption in the System 80+ fire and flood risk analyses. This divisional separation, which is extended also in the SSW/CCW structures, prevents fires and floods from propagating from one division to the other. There are no doors or passageways connecting the divisions of safety-related equipment up to elevation 70+0 ft.

Separate ventilation systems for each division minimize the possibility of smoke, hot gases, and fire suppressants migrating from one division to another.

Electrical separation between the two safety-related divisions is maintained.
All drains are divisionally separated. Drains within a division drain to the lowest level which has adequate volume to collect water from a break in any division. They are sized to handle the potential discharge of fixed fire-suppression systems and fire hoses.

During plant shutdown operation, the integrity of fire and flood barriers between areas in the same division, such as quadrants, where systems comprising the alternate shutdown success paths are located, should be maintained. The COL applicant will maintain configuration control of fire/flood barriers for shutdown operation. This is COL Action Item 19-18. The COL applicant should incorporate in its configuration control program a requirement that at least one quadrant within the subsphere (with at least one SCS/CSS pump operable) be maintained physically isolated with its water-tight fire doors closed during Modes 4, 5, and 6 to help prevent common-mode failures from internal floods or fires.

During plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures, and operator knowledge of plant configuration. The control of these activities is an important COL applicant responsibility.

All fire barriers which provide separation between the two divisions are rated for at least 3 hours. It was assumed that all fire doors and penetrations within the fire barriers are maintained with high reliability during power operation to prevent the propagation of fire from one area to the next.

All flood barriers are capable of withstanding water pressures generated by floods in adjoining areas. All water-tight doors and penetrations are maintained with high reliability during power operation to prevent the propagation of water from one area to the next.

Each half of the subsphere is compartmentalized to separate redundant safe-shutdown components, to the extent practicable while maintaining accessibility requirements. The subsphere, which houses the front-line safety systems is compartmentalized into quadrants, with two quadrants on either side of the divisional structural wall. Flood barriers separate quadrants, and maintain equipment removal capability. EFW pumps are located in separate compartments within the quadrants, and each compartment is protected by flood barriers. Flood barriers also separate electrical equipment from fluid mechanical systems at the lowest elevation within the nuclear annex. Elevated equipment pads prevent equipment from being inundated in the event of flooding.

Flood protection is integrated into the floor drainage systems. The floor drainage systems are separated by division and Safety Class 3 seismic Category 1 valves which prevent backflow of water to areas containing safety-related equipment. Each subsphere quadrant has its own separate sump equipped with redundant Safety Class 3, seismic Category I sump pumps and associated instrumentation. These pumps are also powered from the diesel generators in the event of LOOP. The nuclear annex also has its own divisionally separated floor drainage system, having no common drain lines between divisions. Floors are gently sloped to allow good drainage to the divisional sumps. Floor drains are routed to the lowest elevation to prevent flooding of the upper elevations. The lowest elevation in each division has adequate volume to collect water from a break in any system without flooding the other division. In addition, potential discharge of fixed fire-suppression systems and fire hoses is considered in the sizing of floor drains to preclude flooding of areas should the fire protection systems be initiated.

The COL holder will maintain a well-trained and well-prepared fire brigade.

The System 80+ low-pressure systems which interface with the RCS are protected against ISLOCA by a combination of increases in the piping pressure limits and autoisolation capability based on pressure sensors.

Solid state switching devices and electromechanical relays resistant to relay chatter will be used in the Nuplex 80+ protection and control systems. Use of these devices and relays either eliminates or minimizes the mechanical discontinuities associated with similar devices at operating reactors.

During the construction stage, the COL holder will be able to consider as-built information. The plant-specific PRA developed for the design certification should be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, TS, plant specific EOPs and design changes.

As plant experience data accumulates, failure rates (taken from generic data bases) and human errors assumed in the design PRA are to be updated and incorporated, as appropriate, into the operational reliability assurance process (O-RAP) (see Section 17.3 of this report).

ABB-CE has stated that the COL applicant (COL Action Items 19-5 and 19-12) should provide an updated internal fire PRA that takes into account design details, such as cable routing and door locations as well as fire detection and suppression system location, to search for internal fire vulnerabilities in the detailed design.

ABB-CE has stated that the COL applicant (COL Action Items 19-6 and 19-12) should provide an updated internal flood PRA that takes into account design details, such as pipe routing, door locations, and flood barriers, to search for internal flooding vulnerabilities in the detailed design.

There are 3-hour-rated fire barriers as well as flood barriers between quadrants, and the power and control cables to the quadrants are separated by fire and flood barriers.

The COL applicant should perform a site-specific PRA-based analysis of external flooding, hurricanes, or other external events pertinent to the site to search for site-specific vulnerabilities. This is COL Action Item 19-12. This site-specific PRA should be submitted at the time of the COL application and updated, as necessary, to account for ongoing first of a kind engineering.

Nuclear Annex and Reactor Building

There are no sources of "unlimited" external flooding in the RB. The interface between the CCWS and the ultimate heat sink (through the service water system (SWS)) is located in a separate structure outside the RB.

Consequential flooding of safety-related plant structures from turbine building sources is prevented by the following design features: (1) plant grade below openings to safety-related structures, (2) openings to safety-related structures above the maximum flood level for the turbine building; and (3) site grade such that water would flow away from structures in which safety-related equipment is located.

Drains in the nuclear annex and the RB are divisionally separated, have seismic Category I valves to prevent backflow, and are sized to handle fire-protection system discharges. Each subsphere quadrant has its own redundant seismic Category I sump pumps that can be powered off the EDGs.
The possible sources of internal flooding within the nuclear annex and RB are located below elevation 70+0 ft.

The seals for the underground pipe chase (contains CCW piping) between the nuclear annex and the CCW building will be capable of withstanding an internal flood from a pipe break in the CCWS/SSWS building (e.g., service water).

Chemical and Volume Control System

Divisional separation exists between redundant charging pumps and their power supplies.

There will be diverse RCP seal injection capability using positive displacement pump that is diverse from the CVCS and can be powered from either the EDGs or the CTG. The dedicated seal injection pump (air-cooled positive displacement pump) should be located in such a manner as to minimize its vulnerability to internal floods and fires that could also affect the primary means of providing RCP seal cooling or RCP seal injection.

Instrument Air System (IAS)

Divisional separation exists between redundant trains of instrument air and their power supplies.

Instrumentation and Control

To provide sufficient diversity and defense in depth to mitigate all postulated accidents even assuming a CCF within the plant protection system (PPS), the System 80+ instrumentation and control systems provide the manual hardwired engineered safety feature actuation system (ESFAS) for the controls and, for display, there are hardwired key indications of critical function status for postaccident monitoring.

Rapid Depressurization System

The following are some important aspects of the RDS as represented in the PRA:

One function of the RDS (which is part of the SDS) is to provide a safety-grade means of rapidly depressurizing the

RCS manually from the MCR so that SI can be actuated, when the long-term DHR fails. This is an important feature added to the System 80+ design that helps reduce the failure probability of long-term DHR and plant risk with respect to operating reactor designs.

An important function of the RDS, in addition to providing a primary feed-and-bleed cooling capability in conjunction with the SIS, is to provide the capability to depressurize the RCS during a severe accident to minimize the potential for HPME.

The RDS valves are motor operated and will not reclose on high containment pressure.

The RDS valves fail as-is and, therefore, they are not subject to automatic reclosing upon battery depletion.

ABB-CE will provide emergency operation guidelines (EOGs) guidance on use of RDS for feed-and-bleed cooling. Procedures should be provided for the use of the RDS for depressurization of the RCS during a severe accident.

The SDS valves are qualified for DBA conditions.

The reliability of the RDS is important. The COL will ensure high reliability.

Shutdown Cooling System

The following are some important aspects of the SCS as represented in the PRA:

The SCS has two separate and redundant divisions, each with the heat-removal capacity to cool the RCS to cold-shutdown conditions.

The SCS and CSS pumps are designed to be independent, identical, and functionally interchangeable. Either pump in a division can provide flow to either the CSS header or the SCS heat exchanger.

With the SCS heat exchanger bypass throttle valves failed open, there is adequate flow (3,785 L. per min [1,000 gpm]) through the SCS heat exchanger to achieve cooldown over an extended time period.

During plant shutdown operation, the SCS can be aligned to the IRWST to provide RCS inventory makeup or to perform a feed-and-bleed operation.

Instrumentation and controls in the remote shutdown panel ensure that the SCS functions can be performed even when the MCR cannot be used because of a fire.

The SCS discharge valves are capable of opening with a differential pressure equal to the SCS pump shutoff head. This capability is needed for SCS injection from the IRWST to the RCS following ASC.

The SCS piping outside of containment has an ultimate strength in excess of the normal RCS pressure of 2250 psi.

The SCS pumps can be aligned to take suction from the IRWST. The SCS pumps can also be aligned to discharge to the IRWST via the SCS heat exchangers.

The SCS can be aligned to provide IRWST cooling. This backs up the CSS capability for providing IRWST cooling.

The valve isolating the SCS pump suction from the IRWST is capable of passing flow in either direction. This is required so that the SCS pump can draw suction from the IRWST to back up the appropriate CSS pump and the CSS pump can draw suction from the RCS to back up the SCS pump.

With the SCS pumps aligned to the IRWST, the pumps' net positive suction head (NPSH) is adequate to prevent pump cavitation and failure if the IRWST inventory is saturated.

The SCS pump motor in each division is not powered from the same Class 1E 4.16-kV bus as the CSS pump motor in the same division.

Safety Injection System

The following are some important aspects of the SIS as represented in the PRA:

Four redundant trains are arranged in two divisions so the two SIS divisions are completely physically separated from each other.

Each SIS pump train has an independent suction line connection to the IRWST.

The two SIS divisions are completely physically separated from each other outside containment.

SI for primary feed-and-bleed is an important backup DHR method during shutdown operation. A new TS was added requiring two of the four SIS trains to be available during most shutdown modes.

Instrumentation and controls for trains 3 and 4 are provided at the remote shutdown panel to ensure that the SIS functions can be performed even when the MCR cannot be used due to a fire.

Containment Spray System

The following are important aspects of the CSS as represented in the PRA:

In addition to its design-basis capabilities, the CSS provides the capability to cool the IRWST during accidents requiring feed-and-bleed operation.

The CSS pumps' NPSH is adequate to prevent pump cavitation and failure if the IRWST inventory is saturated.

Electrical Distribution System

The following are some important aspects of the EDS as represented in the PRA.
The EDS includes the following features intended to reduce the frequency of LOOP and SBO events.

- The grid system for System 80+ will include at least two preferred power circuits, each having sufficient capacity. They will be continuously energized and available to provide power to safety-related loads. The two designated offsite power transmission lines shall be designed and routed to minimize, to the extent practicable, the likelihood of their simultaneous failure. These circuits shall be routed to ensure no single event, such as a tower falling or a line breaking, can simultaneously affect both circuits in such a way that neither can be returned to service. The two offsite power circuits shall terminate at two switchyards that are physically separate and electrically independent to the extent practicable.

- The turbine generator system and the associated buses are designed to run back to maintain "hotel" loads on a loss of load. This is an important feature incorporated into the System 80+ design for reducing the frequency of SBO events and should be included in the D-RAP and O-RAP.

- The two EDGs have dedicated 125-V dc batteries (division batteries). Therefore, they can start and load without the emergency channel batteries.

- In addition to the two EDGs, the System 80+ design has an AAC power source. This is a non-safety, seismically robust, CTG which is independent and diverse from the EDGs.

- The two EDGs are physically and electrically isolated from each other.

- High reliability of the onsite emergency ac power sources is important also during shutdown operation. A new TS was added requiring that two of the three onsite power sources (i.e., two EDGs and the CTG) be available during shutdown operation.

The reliability of the two switchyards is an important feature contributing to the reduced System 80+ shutdown risk from LOOP/SBO events, as compared to operating reactor designs. When a switchyard is unavailable for maintenance, the COL applicant should ensure that no activities which could fail the operating switchyard are taking place and no fire sources are present. This is COL Action Item 19-17.

Each of the six independent load group channels and divisions of 125-V dc vital I&C power has a separate and independent Class 1E 125-V battery (two division batteries and four channel batteries). Each battery is sized to supply the continuous emergency load of each own load group for a period of 2 hours. The six independent and separate Class 1E 125-V batteries permit operating the I&C loads associated with the turbine-driven EFW pumps for 8 hours, assuming manual load shedding or the use of a load management program. This enhances the SBO coping capability of the System 80+ design.

Each EDG has a complete and separate fuel oil storage system. The storage system has sufficient fuel to permit EDG operation for no less than 7 days.

Each EDG has two independent air-starting systems.

The COL applicant should develop procedures for manually aligning the AAC power supply (CTG) when one of the two diesel generators is unavailable during a LOOP event. This is COL Action Item 19-19.

Breakers between the PNS and the Class 1E buses will be interlocked so that a PNS bus cannot be aligned to a Class 1E bus that is being powered by an EDG.

Procedures for load shedding will be provided.

The structure that houses the CTG must have an HCLPF value of at least that of the CTG itself, or must be designed in such a manner that failure of this structure following a seismic event up to the HCLPF value of the gas turbine will not affect the availability of the gas turbine to perform its intended function.

Station Service Water System and the Component Cooling Water System

The following are some important aspects of the SSWS and CCWS as represented in the PRA:

Each of these systems (i.e., CCWS and SSWS) has two redundant and separate safety-related divisions with heat dissipation capacity to achieve and maintain safe shutdown. Each division has two pumps. Typically during normal operation one SSWS pump and one CCWS pump in each division are running with the second pump of SSWS and CCWS in standby. The standby pump will automatically start if the running pump in that division trips. This configuration reduces the demand failures of pumps and valves which were found to be significant contributors to risk in current generation plants with standby CCWS/SSWS designs.

The supply and return lines in one division of the SSWS are completely separated from the supply and return lines of the redundant division.

The CCW heat exchanger building and the station service water structure have separating divisional walls designed to protect against interdivisional fires and floods (a fire or flood in one division will not affect the other division, for example, by propagation or by causing failure of the divisional wall). There are no cross-connections between the two closed loops in different divisions of the CCWS. The COL applicant is to provide a site-specific PRA to evaluate the design of these buildings to determine if there are any vulnerabilities to internal floods or fires. These are COL Action Items 19-6, 19-7, and 19-12.

SSWS valves in the supply and return lines are locked in the desired position so that only actuation of the pumps is required to place a division in service.

The COL applicant should evaluate the vulnerability of the SSW intake to tornado-generated debris. This is COL Action Item 19-1.

The ESF actuation signals isolate the non-safety-related portion of the CCWS following an accident condition, except for cooling for the RCPs, IAS compressor coolers, charging pump motor coolers, and charging pump miniflow heat exchangers.

Emergency Feedwater System

The following are some important aspects of the EFWS as represented in the PRA:

The EFWS is a dedicated safety system that has two separate and redundant divisions. Each division has two diverse 100-percent-capacity EFW pumps, one motor operated and one turbine driven. Redundancy, diversity, and separation between divisions are important features reducing the failure probability of the secondary-side heat removal.

The EFW pumps in one division can supply feedwater to the SG in the other division through a pipe having at least two normally closed isolation manual valves installed.

Each EFW storage tank (EFWST) can be supplied by gravity flow from the condensate (water) storage tank (CST). This source is isolated by at least two normally closed isolation valves.

The EFW turbine-driven pump in each division is supplied steam from the SG in its division via a pipe connection located upstream of the MSIV. For SBO sequences, the turbine-driven EFW pumps are the only safety system available for removing decay heat. Their operation, however, requires dc power supplied by batteries. No room cooling or other ac source is required for 8 hours.

ASC, which involves cooling the RCS by opening the ADVs and ensuring that EFW is being delivered to both SGs given failure of SI, has a significant impact on the CDF contribution for small LOCAs and SGTR. Given a small LOCA or SGTR with failure of SI, the SCS can be aligned to provide the injection function if the RCS is depressurized below the SCS pump shutoff head.

ABB-CE will provide EOG guidance to the COL applicant for the use of the EFWS, and the turbine bypass system (TBS) or ADVs for ASC and the alignment of the SCS for injection operation.

In-Containment Refueling Water Storage Tank

The IRWST is an important design feature which helps reduce the System 80+ risk with respect to operating reactor designs. Important characteristics are (1) located inside containment, (2) the CSS and/or SCS can be aligned to cool the IRWST contents using the CSS or SCS heat exchangers, respectively, (3) no valve changeover is required for the recirculation mode of emergency core cooling, (4) IRWST inventory can be made up from the BAST, and (5) in conjunction with remote manual valve operation, provides source of water for flooding the reactor cavity in severe accidents.

Main Control Room

The following are features of the System 80+ MCR which were assumed to minimize risk from fires in the MCR:

- The materials in the MCR panels do not independently support combustion.
- The energy sources coming into the control panels are limited to low-power voltage, thus practically eliminating potential ignition sources within the panels.

- A significant portion of the control and indication signals are interfaced to the main control panel via fiber-optic cables.

In designing the Nuplex 80+ advanced control complex, it is important that no significant new human errors be introduced. To this end, during the MCR verification and validation (V&V) process, the COL applicant should qualitatively confirm that the "findings" from the human factors (V&V) plan (as dispositioned) do not lead to a risk-significant increase in error potential over that represented in the System 80+ PRA HRA. If this is not confirmed, the COL applicant should model the additional risk-significant errors in an updated HRA. This aspect of the validation process is addressed in Section 8.1 of the "Human Factors Engineering Verification and Validation Plan for Nuplex 80+," (Reference 3 of CESSAR-DC Section 18.4).

No water lines are routed above or through the MCR and the computer room. HVAC water lines are in rooms around the MCR that have raised curbs to prevent leakage from entering the MCR.

The MCR has its own dedicated ventilation system. This eliminates the possibility of smoke, hot gases, and fire suppressants, originated in areas outside the MCR, to migrate via the ventilation system to the MCR.

Remote Shutdown Panel

Sufficient instrumentation and controls at the remote shutdown panel bring the plant to safe shutdown in case the MCR must be evacuated. Indication and control are provided for EFW, SCS, ADVs, SIS, RDS, CCWS, and SSWS. Equipment that does not have dedicated instrumentation and controls at the remote shutdown panel can be controlled via the operator's module. This provides the ability to control most plant functions, albeit on a limited basis, from the remote shutdown panel.

An MCR fire will not impact the instrumentation and controls located at the remote shutdown panel, or the equipment which is required to place the plant in cold shutdown, because of the following features of the System 80+ design:

- The MCR and the remote shutdown room are located at different elevations and in different fire areas.
- The MCR ventilation system is different from the ventilation system for the remote shutdown room.
- The stairwells connecting the MCR and the remote shutdown room are pressurized, thus not allowing smoke, hot gases, and fire suppressants to migrate from one room to the other.
- The MCR is continuously pressurized to prevent the entry of smoke, hot gases, dirt, and fire suppressants from other areas.

Startup Feedwater System

The SFWS, a non-safety-related system, can be used to deliver feedwater to the SGs following a reactor trip. The SFWS pump is powered from the PNS bus and can be powered by the CTG. The SFWS pump can be aligned to the CST or the degenerator storage tank. With alignment to either storage facility, the NPSH for the pump is adequate to prevent pump cavitation and failure.

Emergency Containment Spray Backup System

An ECSBS is included in the System 80+ design to provide an onsite pumping source independent of ac power buses, with the capability to supply water to the containment spray header from an external source when the normal CSS is not available, including SBO events.

The ECSBS comprises the following design features: (1) an 20 cm (8-in.) diameter "tee" connection to the containment spray recirculation line, (2) an extension of 20 cm (8-in.) diameter Class 2 piping from the tee connection to the exterior of the nuclear annex, (3) external connections for temporary hookup of an external source of water located at or near grade, (4) a portable, onsite pumping source (e.g., fire truck) with the capability to supply sufficient flow against maximum containment pressure to maintain containment pressure below ASME Service Level C limits, and (5) pre-staging of all necessary hoses, fittings, and spool pieces.

The specific flow rate for the pumping device will be developed by the COL applicant as part of the detailed design. This is COL Action Item 19-10.

The detailed ECSBS design and location of all associated valves and connections should take into account expected radiation levels and shielding requirements for any required local operator actions. This is COL Action Item 19-11.

Detailed procedures for use of the ECSBS will be developed by the COL applicant. This is COL Action Item 19-16.

Hydrogen Mitigation System

There is an HMS utilizing igniters to control hydrogen during a severe accident.

The accident management procedures will address use of the HMS. This is part of COL Action Item 19-16.

The hydrogen purge vent to the annulus is not credited in the PRA. However, the use of this vent could decrease the late CCFP.

Reactor Cavity Flood System

A reactor cavity flood system is provided to enhance the coolability of ex-vessel core debris.

Procedures for use of the CFS during a severe accident will be developed by the COL applicant as part of its plant-specific severe-accident management guidelines. This is COL Action Item 19-16.

The reliability of the CFS and associated valves is important. The COL applicant will ensure the reliability of the CFS by inclusion of the system in the reliability assurance program.

Containment Penetrations

The major containment penetrations (equipment hatch, personnel airlocks, and fuel transfer tube) will be designed to ensure that they will maintain leak-tightness up to ASME Service Level C for the containment shell.

Penetrations will be designed to ensure that the selected seal and mounting will provide a minimum of 1 days' containment integrity. This will be achieved by a combination of selecting high-quality and high-capability seals, protectively mounting the seal so that it is not directly exposed to the containment environment, and providing double seals (inner and outer) wherever possible.

The major containment penetrations will be designed to ensure that they will maintain leak-tightness up to the ultimate pressure capacity of the containment shell.

Seal materials in the major containment penetrations will be selected to minimize the potential for overtemperature failure.

Steam Supply System

The reliability of the MSSVs, ADVs, and MSIVs is important. The COL applicant will ensure the reliability of these components.

19.1.8 Conclusions and Findings

The NRC has evaluated the quality of the System 80+ design PRA and its use in the design development and certification process. The NRC concludes that the quality and completeness of the System 80+ PRA is adequate for its intended purposes, such as supporting the design and design certification process. The approaches used by ABB-CE for both the core damage and containment analyses are logical and sufficient to achieve the desired goals of describing and quantifying potential core-damage scenarios and containment performance during severe accidents. The NRC concludes that the use of PRA in the design process helped introduce improved or unique "evolutionary" features (such as the RD capability of the SDS, the IRWST, and the reactor CFS) which contributed to the reduced CDF and CCFP estimates of the System 80+ design when compared with operating PWRs. PRA results and insights were used to identify areas in which it is particularly important to implement the design and operational requirements assumed for design certification (e.g., ITAACs, D-RAP, O-RAP, TS, operator training and procedures). The NRC finds that the System 80+ design represents an improvement in safety over operating PWRs in the United States.

Based on the staff's review of the System 80+ PRA and System 80+ design as set forth in this section (19.1) of this report, the staff finds that the System 80+ design and the submittals made for the System 80+ in the CESSAR-DC meet the intent of the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, dated August 8, 1985, the requirement of 10 CFR 50.34(f)(1)(i) to perform a plant-specific PRA that seeks improvement in the reliability of core and containment heat removal systems, the staff's proposed applicable regulation for analysis of external events for the System 80+ PRA, and the requirement of 10 CFR 52.47(a)(1)(v) for an evolutionary plant design vendor to submit a PRA.

19.2 Severe-Accident Performance

19.2.1 Introduction

The purpose of Section 19.2 is (1) to consolidate the NRC's approach to resolution of severe-accident issues for ALWRs as specified in SECY-90-016, SECY-91-262, SECY-93-087, and the corresponding SRMs, and (2) to evaluate the approach proposed by ABB-CE for resolution of severe-accident issues for the System 80+ design.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and, if accidents should occur, to mitigate their consequences. Siting in less populated areas is emphasized. Furthermore, the NRC, State, and local governments mandate emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population.

The reactor and containment systems design are a vital link in the defense-in-depth philosophy. Current reactors and containments are designed to withstand a LOCA and to comply with the siting criteria of 10 CFR Part 100 and general design criteria of 10 CFR Part 50 (Appendix A). The large-break LOCA and other accidents analyzed in accordance with the NRC's Standard Review Plan (SRP) (NUREG-0800) and documented in Chapter 15 of the CESSAR-DC are commonly referred to as "design-basis accidents" for nuclear power plants.

The high-level of confidence in defense-in-depth approach results, in part, from stringent requirements for meeting single failure criterion, redundancy, diversity, quality assurance, and utilization of conservative models. The staff concludes that existing requirements ensure a safe containment design.

The NRC also has requirements to address conditions beyond the traditional design-basis spectrum, such as ATWS (10 CFR 50.62), SBO (10 CFR 50.63), and combustible gas control (10 CFR 50.44); however, a definitive set of regulatory requirements for addressing specific severe-accident phenomena does not exist. Existing regulations which require conservative analyses and inclusion of features for design-basis events provide margin for severe-accident challenges. This design-basis margin coupled with regulatory guidance to address severe accidents in the form of policy positions provides a robust design that satisfies the Commission's policy statement on severe accidents.

In an SRM, dated January 28, 1992, on SECY-91-262, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues. The effect of these actions on the System 80+ is that the criteria specified for resolution of severe accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the System 80+ design certification rulemaking as applicable regulations. The following discussion describes the criteria that were used for the deterministic evaluation of severe accident issues.

19.2.2 Deterministic Assessment of Severe Accident Prevention

19.2.2.1 Severe-Accident Preventive Features

The System 80+ is designed to cope with plant transients and LOCAs without any adverse impact on the environment. However, the potential does exist, albeit remote, for a LOCA or seemingly ordinary plant transient coupled with numerous plant safety system failures to progress to a severe accident with the potential for substantial offsite releases.

Accident initiators can be separated into two general groups — transients and LOCAs. Transients include planned reactor shutdowns and transients which result in reactor scrams. Examples of transients are manual shutdown, steamline break (SLB), SGTR (also a form of LOCA), LOOP, and loss of feedwater (LOFW). In addition to these transients, there is an entire spectrum of LOCAs which are accident initiators. LOCAs fall within three categories: small, medium, and large, based on the size of the line break.

Following the accident initiator, normal and emergency plant systems respond to control reactivity, reactor pressure, reactor water level, and containment parameters within the design-bases spectrum. Of most importance is to ensure inventory control and sufficient heat removal from the core to prevent overheating and subsequent fuel damage. Failure to provide heat removal or inventory control; results in core uncovery, fuel overheating, and the potential for oxidation and melting of the reactor core.

In response to accident initiators identified through operating reactor experience and performance of probabilistic risk assessment, the NRC developed criteria for evolutionary LWRs to prevent the occurrence of such initiators from leading to a severe accident. These criteria were specified in SECY-90-016 and SECY-93-087 and include design provisions for the following: ATWS, SBO, fires, and ISLOCAs.

19.2.2.1.1 Anticipated Transient Without Scram

An ATWS is an anticipated operational occurrence (AOO) followed by the failure of the trip portion of the RPS. AOOs are those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Dependent upon the transient and its severity, the plant may recover and continue normal operation or the plant may require an automatic shutdown (scram) via the RPS. The RPS is designed to safely shut down the reactor to prevent core damage.

These transients when coupled with a failure of the RPS may lead to conditions beyond the design basis of the plant. In these cases, the reactor must be manually scrammed in order to avoid reactor fuel damage or coolant system damage. Subsequent failure of the manual scram system and inadequate core cooling (ICC) would lead to core damage.

Transients with the greatest potential for significant damage to the reactor core and containment are those which lead to an increase in reactor pressure and temperature, a loss of heat sink, or a failure of the RPS to scram the reactor. During an ATWS event, reactor power, pressure, and temperature must be controlled or the potential exists for a severe accident.

The ATWS rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. For PWRs, the ATWS rule specifies inclusion of a diverse scram system from the sensor output to interruption of power to the control rods. In Sections 7.7 and 15.3.10 of this report, the NRC concludes that the System 80+ design complies with the ATWS rule.

19.2.2.1.1.1 Features To Prevent and/or Mitigate

In SECY-90-016, the staff recommended that the Commission approve the staff's position that diverse scram systems should be provided for evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position, but directed that if an applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

The System 80+ design includes a control-grade alternate protection system to provide an alternate reactor trip signal and an alternate EFW actuation signal separate and diverse from the safety-grade reactor trip system.

19.2.2.1.1.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that evolutionary LWR designs should have diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The System 80+ design has a diverse alternate scram system. The System 80+ design complies with the ATWS rule, as concluded in Sections 7.7 and 15.3.10 of this report, and the design is capable of satisfactorily mitigating the effects of an ATWS and preventing an ATWS event from evolving into a severe accident with core damage. The staff concludes that the System 80+ design conforms to the criteria specified in SECY-90-016 by incorporating the features discussed above.

19.2.2.1.2 Station Blackout

An SBO involves the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by AAC sources, nor does it assume a concurrent single failure or DBA.

During normal plant operation, power is supplied to the Class 1E distribution system from the main generator. After plant shutdown, the preferred power source is the offsite grid, which provides a continuous source of ac electric power to equipment required to maintain core coolability. If the power from the offsite grid is not available, the onsite distribution system will sense an undervoltage condition and initiate a transfer to the EDGs for continued power. In the event of the loss of both the offsite grid and EDGs, an SBO has occurred. Because most DHR and containment heat removal systems are dependent upon ac power, the loss of ac power could lead to increases in temperature and pressure and subsequent failure to provide core cooling may lead to a severe accident.

The SBO rule (10 CFR 50.63) allows several design alternatives to ensure that a plant is able to withstand an SBO for a specified duration and recover. A complete evaluation of the System 80+ relative to the SBO rule is in Section 8.5 of this report.

19.2.2.1.2.1 Features To Prevent and/or Mitigate

In SECY-90-016, the staff concluded that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full capacity) AAC source of diverse design that is consistent with the guidance in Regulatory Guide (RG) 1.155, "Station Blackout," August 1988, and is capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve imposition of an AAC source for evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

The System 80+ design has two independent electrical divisions, each with high- and low-pressure water injection capability, each powered by a full-capacity EDG, and each division capable of independently shutting down the reactor. In addition, the System 80+ design includes non-safety-grade CTG (AAC source) to back up the EDGs. Two division and four channel batteries provide an SBO coping capability assuming manual load shedding or the use of a load management program. This permits operating the instrumentation and control loads associated with the turbine-driven EFW pumps for 8 hours.

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine generator system and the associated buses are designed to run back to maintain hotel loads on a loss of grid. These features aim at reducing the frequency of LOOP initiating events and, therefore, the frequency of SBO.

19.2.2.1.2.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that designers should meet the SBO rule by including an AAC power source (i.e., CTG) of diverse design capable of powering at least one complete set of normal shutdown loads. The System 80+ design complies with this request by including a CTG with this capability. The System 80+ also has the capability to survive at least an 8-hour blackout period. The staff concludes that the System 80+ has fully conformed to the criteria of SECY-90-016.

19.2.2.1.3 Fire Protection

The Commission concluded that fire-protection issues that have been raised through operating experience and the External Events Program must be resolved for evolutionary LWRs. In SECY-90-016, the staff recommended that current NRC guidance to resolve fire-protection issues be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents and DBAs. As indicated in SECY-90-016, the System 80+ design must ensure that safe shutdown can be achieved, assuming that all equipment in any area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the MCR is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the MCR is included in the design. The System 80+ design must also provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practical, that one shutdown division will be free of fire damage. Additionally, the System 80+ design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that they could impair safe-shutdown capabilities, including operator actions. Fire protection is further discussed in Chapter 9 of this report.

19.2.2.1.3.1 Features To Prevent and/or Mitigate

In CESSAR-DC Section 9.5.1, "Fire Protection System," ABB-CE describes and evaluates the System 80+ features provided to prevent and mitigate fires. In particular, this section addresses protection of safe-shutdown equipment, passive fire-protection features, fire detection, fire-protection water supply system, water fire-suppression systems, gaseous fire-suppression systems, fire extinguishers, emergency communication and lighting, emergency breathing air, curbs and drains, smoke control, access/egress routes, construction materials and combustible contents, and interaction with other systems.

19.2.2.1.3.2 Basis for Acceptability

On the basis of the staff's evaluation of CESSAR-DC Section 9.5.1 and the discussion above and documented in Section 9.5.1 of this report, the staff concludes that the System 80+ design complies with the criteria in SECY-90-016 and is acceptable for preventing and mitigating threats from fires for severe accidents.

19.2.2.1.4 Interfacing Systems Loss-of-Coolant Accident

ISLOCAs are defined as a class of LOCAs in which the RCS pressure boundary, interfacing with a system of lower design pressure, is breached. The breach may occur in portions of piping located outside of the primary containment, causing a direct and potentially unisolable discharge from the RCS to the environment. An ISLOCA is of concern because of potential direct releases to the environment, loss of core cooling, and loss of core makeup.

High/low-pressure interfaces occur on many lines including the SDC heat exchangers or the containment spray pumps which are capable of substituting for the SDC pumps. An ISLOCA occurs when high pressure is introduced to a low-pressure system due to a valve(s) failure or an inadvertent valve actuation. In either case, the overpressurization can cause the low-pressure system or components to fail.

19.2.2.1.4.1 Features To Prevent and/or Mitigate

In SECY-90-016, the staff recommended that evolutionary LWR designs reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the full RCS pressure. The "extent practicable" phrase is a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to an URS equal to full RCS pressure. The staff further recommended that systems that have not been designed to withstand full RCS pressure should include (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the MCR when isolation valve operators are de-energized, and (3) high-pressure alarms to warn MCR operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In its June 26, 1990, SRM, the Commission approved the staff's position on ISLOCA provided that all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

ABB-CE performed a systematic evaluation of interfacing systems to ensure that the SECY-90-016 requirements are satisfied. The resolution of this issue is discussed in Section 20.2 of this report.

19.2.2.1.4.2 Basis for Acceptability

Generic Safety Issue 105: ISLOCA is addressed in Section 20.2 of this report. As indicated there, the staff concludes that the System 80+ design complies with the criteria from SECY-90-016.

19.2.3 Deterministic Assessment of Severe-Accident Mitigation

19.2.3.1 Overview of the System 80+ Containment Design

The System 80+ primary containment design consists of a 61 m (2000-ft)-diameter spherical steel shell with a nominal wall thickness of 4.45 cm (1.75 in.). This wall will be reinforced around primary containment penetrations to structurally compensate for these openings. The primary containment encloses the nuclear steam supply system, the IRWST, SITs, the refueling canal, and associated mechanical, electrical, and HVAC support components. The spherical steel shell below the reactor cavity is protected by a minimum of 0.9 m (3 ft) of concrete with an additional 5.5 m (15 ft) of concrete below the steel shell.

The primary containment has 94,600 m^3 (3,340,000 ft^3) of net-free volume, and its internal structures are arranged in a manner to promote mixing throughout the containment atmosphere and accommodate the pressurization from condensible and non-condensible gas releases during severe accidents. The internal structures surrounding the SGs are especially effective at promoting natural circulation within containment by causing a "chimney" effect. The containment contains 80 igniters to control hydrogen generated during severe accidents.

The primary containment is totally enclosed by a shield building made of reinforced concrete. The containment shield building is designed to provide biological shielding and external missile protection for the containment vessel and safety-related equipment. In addition, the annulus ventilation and filtration system provides a mechanism for reducing fission-product releases following severe accidents.

Steam from a reactor depressurization event is condensed in the IRWST. The IRWST is the primary heat sink and may be cooled by either the SCS or the CSS heat exchangers. Either system supplies water at about 1,480 kPa (200 psig) discharge pressure. The System 80+ design does not rely on low pressure SI; however, the SCS delivery pressure is sufficient to function as a low-pressure emergency core cooling system (ECCS) and as a surrogate for the SIS. Typically low-pressure ECCSs in operating PWRs deliver water between 1,720 kPa and 3,450 kPa (250 and 500 psi). The depressurization system is expected to reduce primary system pressure to approximately 1,140 kPa (150 psig). The IRWST supplies water to the CFS which provides a means of flooding the reactor cavity during a severe accident, for the purpose of cooling the core debris in the reactor cavity and scrubbing fission-product releases. The water flows first into the HVT through four 30-cm (12-in.)-diameter MOVs and then into the reactor cavity through two 25-cm (10-in.)-diameter MOVs. The IRWST also supplies water to the CSS which can be used to reduce containment temperature and pressure and remove iodine from the containment atmosphere following severe accidents.

The spherical steel containment can be vented, in the case of an internal pressurization that may challenge containment integrity, through two 8-cm (3-in.)-diameter hydrogen purge vents. ABB-CE has provided this venting capability; however, ABB-CE has demonstrated that venting is not needed for most of the severe-accident events. For those sequences in which venting would aid in limiting the containment pressure below ASME Service Level C, venting would not be needed before 24 hours into the event. The use of the hydrogen purge vent for containment pressure control is the responsibility of the technical support center. ABB-CE used the ASME Boiler and Pressure Vessel Code to determine the containment pressure that may be reached without exceeding ASME Service Level C. ASME Service Level C loading conditions allow material strains representative of incipient yield, assuming minimum material properties, and consequently provides a conservative estimate of the containment ultimate capacity. The pressure limits determined in accordance with ASME Service Level C criteria decrease from about 1.00 MPa (145 psia) at an average steel shell temperature of 143 °C (290 °F) to 0.930 MPa (135 psia) at a temperature of 232 °C (450 °F).

19.2.3.2 Severe-Accident Progression

The processes, both physical and chemical, that may occur during the progression of a severe accident, and how these phenomena affect containment performance, are described in this section. This description is intended to be generic in nature; however, many aspects of severe-accident phenomena depend on the specific reactor type or on the containment design features. This information has been extracted from NUREG/CR-5132, "Severe Accident Insights Report," April 1988, NUREG/CR-5597, "In-Vessel Zircaloy Oxidation/ Hydrogen Generation Behavior During Severe Accidents," September 1990, and NUREG/CR-5564, "Core-Concrete Interactions Using Molten UO_2 With Zirconium on a Basaltic Basemat," August 1992.

Severe accident progression can be divided into two phases, an in-vessel stage and an ex-vessel stage. The in-vessel stage generally begins with insufficient DHR and can lead to melt-through of the reactor vessel. The ex-vessel stage involves the release of the core debris from the reactor vessel into the containment and such resulting phenomena as CCI, FCI, and DCH.

19.2.3.2.1 In-Vessel Melt Progression

In severe accidents that proceed to vessel failure and release of molten core material into the containment, the in-vessel melt progression establishes the initial conditions for assessment of the thermal and mechanical loads that may ultimately threaten the integrity of the containment. In-vessel melt progression encompasses the phenomena and processes involved in a severe core- damage accident starting with core uncovery and initial heatup, and continuing until either (1) the degraded core is stabilized and cooled within the reactor vessel or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes in the System 80+ that can occur during in-vessel melt progression include

- core heatup resulting from loss of adequate cooling
- metal-water reaction and cladding oxidation
- eutectic interactions between core materials
- melting and relocation of cladding, structural materials, and fuel
- formation of blockages near the bottom of the core due to the solidification of relocating molten materials (wet core scenario)
- drainage of molten materials to the vessel lower head region (dry core scenario)
- formation of melt pool, natural circulation heat transfer, crust formation, and crust failure (wet core scenario)
- lower head breach resulting from failure of a penetration, or from local or global creep-rupture

Decay heat produced by the core must be removed to achieve adequate core cooling. Adequate core cooling can be accomplished in the System 80+ by either providing enough cooling water flow to the reactor core and by removing the decay heat through the SGs or via feed-and-bleed. The mechanisms by which decay heat is removed from the reactor core include a four-train SIS with DVI, functionally interchangeable SDC and CSS, and SDS. Cooling water flow to the SGs is provided by the main and EFWS. If the decay heat is transferred to the containment from the core, it can be removed by containment heat removal systems such as the CSS and the CCWS.

In the event that all safety and non-safety systems fail to remove the decay heat, the core will heat up to the point at which damage to the fuel and fuel cladding may occur. Decay heat is transferred through the radiative, conductive, and convective heat transfer to the steam, other core materials, and non-fuel materials within the reactor. The insufficient cooling supply results in coolant boiloff and a decreasing level within the reactor vessel as the decay heat generation exceeds the heat removal rate. The coolant level within the core further decreases so that the fuel rods above the coolant level are only cooled by rising steam. The fuel rods begin to overheat and cladding oxidation in the presence of steam begins at high temperatures. As the cladding oxidizes in the presence of steam, hydrogen, and additional heat are generated. The fuel cladding is made of a zirconium alloy called Zircaloy.

The initial Zircaloy oxidation involves oxygen diffusion through a ZrO_2 surface layer. As the fuel rods continue to heat up from decay heat and exothermic zirconium oxidation reaction occurs, the materials within the reactor with low melting points are expected to melt first and may form eutectics. Eutectics are mixtures of materials with a melting point lower than that of any other combination of the same components.

Zircaloy, with a melting point of 1,757 °C (3,194 °F), begins to melt, breaking down the protective ZrO_2 layer which exposes unoxidized Zircaloy. Following this, local melting of the fuel rods may cause changes in the core geometry resulting in differing steam flow paths. This can lead to an increase in the oxidation process as access to the unoxidized Zircaloy is made available; on the other hand, the melt formation or changes in the steam flow path could reduce the Zircaloy surface available for oxidation and thereby decrease the overall reaction process. In some accident scenarios in which residual amounts of water remain in the bottom of the core and lower plenum, substantial steaming and oxidation can take place.

In addition to oxidation, the potential exists for the Zircaloy to interact with the UO_2 fuel, forming low-melting-point eutectics. Formation of eutectics may decrease the effective surface area for oxidation and overall oxidation rate. The melting point of Zircaloy is dependent upon its state and lattice structure. It has three melting points which include 1,877 °C (3,410 °F) (beta-Zr), 1,977 °C (3,590 °F) (alpha-Zr(O)), and 2,677 °C (4,850 °F) (ZrO_2). When partially oxidized Zircaloy is in contact with UO_2, an alpha-Zr(O)/UO_2-based eutectic will form with a liquefaction temperature of approximately 1,897 °C (3,446 °F). Therefore, in the presence of good fuel/cladding contact, fuel liquefaction and melt relocation will commence around this temperature. This has the potential to affect the oxidation behavior of Zircaloy-based melt.

Various severe fuel damage (SFD) test programs sponsored by the NRC indicate that oxidation of the Zircaloy is largely controlled by the availability of a steam supply and that high rates of hydrogen generation can continue after melt formation and relocation. Some of these experiments indicate that the majority of the hydrogen was generated after onset of Zircaloy melting and fuel dissolution. In steam-rich experiments, oxidation took place over most of the fuel bundle length and most of the hydrogen is generated early. For steam-starved experiments, oxidation was limited to local regions of the fuel bundle and the majority of the hydrogen is generated after the onset of Zr/UO_2 liquefaction and relocation.

The System 80+ design contains more than 25,240 kg (55,655 lb) of zirconium in the active fuel region which has the potential to generate more than 1,100 kg (2,440 lb) of hydrogen. Hydrogen production and accumulation may represent challenges to the containment in numerous ways, including deflagration, detonation, and pressurization, as hydrogen gas is non-condensible. The System 80+ containment has 80 hydrogen igniters to consume hydrogen as it is produced during a severe accident. Because of the large containment volume, System 80+ is not threatened from pressurization of the containment from generation of hydrogen gases. The resulting pressures are well below ASME Code Service Level C limits.

The SFD tests indicated the potential for incoherent melt-relocation due to non-coherent temperatures within the test bundles. This is because of the different core materials present with a wide range of melting points and eutectic temperatures. Formation of eutectics would result in a nonuniform melting and relocation process. Further differences in the melt-relocation process can be attributed to asymmetric bundle heating which can increase upon Zircaloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher temperature Zircaloy will consume the available steam through oxidation at a quicker rate. The oxidation reaction increases the hotter areas to even higher temperatures, which further increases the oxidation rate and the local temperatures. This autocatalytic nature of Zircaloy oxidation appears to contribute to asymmetric bundle heatup and the potential for incoherent melt relocation behavior.

As the temperature of the core increases, the fission products are vaporized and released. These fission products are then carried by steam and/or hydrogen throughout the primary system and are subject to deposition on the surfaces of internal components. The deposition mechanisms include condensation, gravitational settling, and thermophoresis. The fission products that are not deposited remain airborne and are released to the containment, where the dominant removal mechanisms are gravitational settling and, potentially, diffusiophoresis.

The core melt progression, including relocation and fission product release, becomes increasingly difficult to predict as it continues to degrade. The core melt could relocate into the lower reactor vessel plenum. If water is present in the lower plenum, the potential exists for in-vessel steam explosions, where molten core rapidly fragments and transfers its energy causing RSG and shock waves. On the other hand, the core debris within the lower plenum may quickly melt through the reactor vessel or interact with available water before melting through and entering the reactor cavity.

The in-vessel core melt progression, including core degradation, relocation, and failure of the reactor vessel, contains considerable uncertainty. This uncertainty includes

- the potential for in-vessel steam explosion
- the interaction between core debris and internal vessel structures
- the time and mode of vessel failure
- the composition of the core debris released at vessel failure
- the amount of in-vessel hydrogen generation
- the in-vessel fission-product release and transport
- retention of fission products and other core materials in the RCS

19.2.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure, the primary system pressure at reactor vessel failure, the composition, amount, and character of the molten core debris expelled, the type of concrete used in containment construction, and the availability of water to the reactor cavity. The initial response of the containment from ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented by design features, early containment failure mechanisms and bypass usually dominate risk consequences. Early containment failure mechanisms result from energetic severe-accident phenomena such as HPME with DCH and EVSEs. The long-term response of the containment from ex-vessel severe-accident progression is largely a function of the containment pressure and temperature due to CCI and the availability of mechanisms to remove heat from the containment.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form, causing fragmentation into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the reactor cavity and into the upper containment. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. Direct attack on the steel shell is precluded in the System 80+ design because the steel shell is either protected by concrete or by the crane wall. This severe-accident phenomenon is known as HPME with DCH. To prevent this phenomenon, the System 80+ design has incorporated a reliable RDS to provide assurance, that in the event of a core-melt scenario, the reactor vessel would fail at a low RCS pressure. Should the reactor vessel fail at a high pressure, the design of the System 80+ containment provides an indirect pathway from the reactor cavity to the upper compartments of the containment in an effort to decrease the amount of core debris that could contribute to DCH.

Reactor vessel failure at high or low pressure coincident with water present within the reactor cavity may lead to interactions between fuel and coolant with the potential for RSG or steam explosions. RSG involves the pressurization of containment compartments from nonexplosive steam generation beyond the capability of the containment to relieve the pressure so that the containment fails because of local overpressurization. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. ABB-CE concludes that the System 80+ plant is capable of withstanding the loads from the most likely FCI.

The eventual contact of molten core debris with concrete in the reactor cavity will lead to CCI. CCI involves the decomposition of concrete from core debris and can challenge the containment in various mechanisms, including (1) pressurization due to the production of steam and noncondensible gases to the point of containment rupture, (2) the transport of high-temperature gases and aerosols into the containment leading to high-temperature failure of the containment seals and penetrations, (3) containment liner melt-through, (4) reactor support structures melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors, including the availability of water to the reactor cavity, the containment geometry, the composition and amount of core melt, the core-melt superheat, and the type of concrete involved.

The System 80+ design has several design features to mitigate the effects of CCI. These include a CFS that can be supplied from the IRWST or externally through the CSS, and limestone-based concrete for the reactor cavity floor. The CFS has been designed to provide water to assist in the cooling of core debris before it enters the reactor cavity. The CSS is capable of providing water, from the IRWST or through an external source, to control containment pressurization. Water entering the containment from the CSS will gather in the HVT where it can be directed to the reactor cavity by the CFS, thereby providing an external means of flooding the reactor cavity. The limestone-based concrete protects the containment liner from melt-through.

19.2.3.3 Severe-Accident Mitigative Features

19.2.3.3.1 Hydrogen Generation and Control

Generation and combustion of large quantities of hydrogen is a severe-accident phenomenon that can threaten containment integrity. The major source of the hydrogen generated is from the oxidation of zirconium metal with steam when the zirconium reaches temperatures well above normal operating levels. This reaction is commonly referred to as the metal-water reaction.

Research indicates that in-vessel hydrogen generation associated with core-damage can vary over a wide range. The specific amount of oxidation is dependent on a variety of parameters related to sequence progression. These include the RCS pressure, the timing and flow rate of reflooding if it occurs, and the temperature profile of the reactor core during the course of the accident sequence. In addition, ex-vessel hydrogen generation must be considered. Hydrogen is produced as a result of ex-vessel core debris reacting with steam or concrete, or both.

19.2.3.3.1.1 Features to Prevent and Mitigate

In 10 CFR 52.47(a)(1)(ii), the NRC requires applicants for a standard design certification to demonstrate compliance with any technically relevant portions of the TMI requirements in 10 CFR 50.34(f). In 10 CFR 50.34(f)(2)(ix), the NRC requires a system for hydrogen control that can show with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an amount of hydrogen equivalent to the amount that would be generated from a 100-percent fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

In SECY-90-016, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

The System 80+ core contains approximately 25,240 kg (55,655 lb) of zirconium in the active fuel-clad region. Oxidation of this amount of zirconium with steam would produce about 1,100 kg (2,440 lb) of hydrogen. This amount of hydrogen uniformly distributed throughout the containment would result in a hydrogen concentration of approximately 13 percent. Therefore, to comply with 10 CFR 50.34(f)(2)(ix), ABB-CE has equipped the System 80+ with a HMS composed of 80 shielded GMAC model 7G thermal igniter glow plugs. The intent of the HMS is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture. This early combustion will limit the hydrogen concentration well below the 10-percent limit referenced in the rule.

The efficiency of the GMAC model 7G thermal igniter has been investigated in several experimental programs such as the program of the Nevada test site, the Hydrogen Igniter Experimental Program at Lawrence Livermore National Laboratory, and tests conducted by Fenwal, Incorporated. These programs showed that the glow plug could effectively ignite hydrogen mixtures as low as 6 percent by volume and that ignition above 8 percent by volume of hydrogen consistently resulted in complete combustion. The Fenwal tests indicated that upward burns would propagate at hydrogen concentrations as low as 4 percent by volume; at 6.5 percent by volume the burn will propagate sideways and at 8.5 percent by volume the burn will propagate in all directions. The results of various experimental programs conducted at these and other facilities are summarized in NUREG/CR-5079, "Experimental Results Pertaining to the Performance of Thermal Igniters" (October 1989).

Tests of these igniters were conducted to support licensing of operating reactors with ice condenser containments and Mark III containments. ABB-CE has shown that this data base is directly applicable to the System 80+ design. Therefore, it is the staff's position that, properly placed and powered, the GMAC model 7G thermal igniter can maintain uniformly distributed hydrogen concentrations below 10 percent.

In order to ensure a highly reliable HMS, two igniters have been supplied in each subvolume in addition to adding igniters in the large upper region of the containment. A subvolume is defined as a region which has some level of air flow restriction. The redundant igniters have been divided equally into two redundant groups, A and B.

Particular attention has been paid to providing a reliable power source to the igniters for all possible conditions. All 80 igniters are capable of being powered via offsite power and the emergency diesels. This is the same as all operating plants that have igniters. However, ABB-CE has provided two additional sources to ensure that power is available at all times to the igniters. The third source is a CTG and is considered by ABB-CE to be the primary backup to the EDGs. The fourth source is from batteries. In case of SBO sequences, 34 igniters can also be supplied power for a minimum of 4 hours by the Class 1E division batteries. The HMS components are non-nuclear safety-related, since they are not required to prevent or mitigate the consequences of a DBA, but are designed to sustain seismic Category I loads.

The staff's evaluation of ABB-CE's placement of the igniters was quite involved, but always came down to the critical question, "If one could develop a sequence in which hydrogen could either be generated in or pass through the volume in question, then igniters should be provided." The starting point was the consideration of detailed three dimensional drawings of the System 80+ layout. Additionally, a series of confirmatory analyses were performed by ABB-CE using the MAAP 4 generalized containment model which were confirmed by the staff using CONTAIN. Results from these analyses assisted ABB-CE in the placement of igniters within the system.

In combination with the analyses, one also considered the possible sources of hydrogen. Two entry points were considered possible. Hydrogen generated in-vessel during a LOCA would be released directly into the containment through the line rupture. Therefore, igniters were placed above and in the vicinity of all RCS primary piping and non-isolable connecting piping to account for these hydrogen sources. The other and more dominant pathway is associated with all transients with an in-tact primary system or small-break LOCAs. These conditions would direct hydrogen to the IRWST via the SDS. Inside the IRWST, four igniters are located in the freeboard space above the spargers.

Any hydrogen not burned in the IRWST, because of steam inerting or a lack of oxygen to support combustion, would flow out of the IRWST through 18.6 m^2 (200 ft^2) of vent area. The vents are located at the bottom of the SG compartment on the El. 91 ft 9 in. and are inside the wing walls. Once inside the SG compartment, the hydrogen will be burned by three sets of four igniters located at Els. 100 ft, 126 ft, and 164 ft.

This briefly summarizes the process that was used to develop the placement and number of igniters in the System 80+ containment. This process is summarized in Table 5.2-1 of CESSAR-DC Appendix 19.11K, titled, "Summary of Specific Igniter Placement and Design Criteria for System 80+," and it lists the criteria used by ABB-CE to determine the 40 igniter regions. Some of the other critical criteria for locating the hydrogen igniters are listed below:

- All System 80+ enclosures, which are vented, have been supplied with a pair of igniters.

- All igniters have been placed approximately 3 m (10 ft) below solid surfaces, such as ceilings, to promote upward burning.

- Igniters have been located away from equipment and instrumentation required during and after a severe accident, or necessary radiative shielding will be provided.

- With the exception of the dome region, igniters in the flowpaths will cover a volume of less than 1,416 m^3 (50,000 ft^3).

- Igniters positioned in the vicinity of the containment shell will be reviewed by the COL to ensure that local heating effects are negligible. If local heating of the shell is a concern, appropriate radiative shielding will be installed between the ignition source and the shell.

The staff has reviewed the entire process and believes that adequate igniter coverage has been provided.

Although placement and number of igniters appears to be adequate, the staff was concerned about the effectiveness of the igniters when the containment comes from an inerted condition into a flammable condition. During a severe accident, the containment atmosphere may be inerted, due to high steam concentrations, thus the igniters would not be able to burn the hydrogen. If this atmosphere were to experience rapid condensation, due to initiation of the CSS, a potentially detonable mixture could be formed before the igniters would ignite the mixture.

Testing done at Whiteshell by Atomic Energy of Canada Ltd. (AECL) and documented in EPRI NP-5254, July 1987, addressed the ignition of mixtures in steam condensing atmospheres using the GMAC 7G thermal glow plug. Mixtures initially steam inerted were ignited as the mixture passed into the flammable region. The condensation took place over a 20- to 30-minute period. Although these experiments are not prototypical of the System 80+, they are believed to be generally applicable as long as the condensation occurs on the order of several minutes as opposed to several seconds.

Experiments are currently being performed by Sandia National Laboratories (SNL) in the modified Surtsey facility to examine hydrogen igniter performance in a steam condensing atmosphere. A steam condensation test, including a spray, was conducted on November 19, 1993. The test showed that the rate of steam condensation occurs on the order of minutes, not seconds, and that predictions using the CONTAIN computer model were in reasonable agreement with the data. Experiments adding hydrogen and igniters to this environment are expected to be carried out in early 1994.

The conditions in the Surtsey facility are similar to those in the System 80+ containment. The initial results of the test and the supporting analytical runs were transmitted to the staff, in a letter from SNL, dated December 13, 1993. In a letter to the staff, dated January 7, 1994, SNL stated that CONTAIN calculations indicate the water spray drops attain thermal equilibrium relatively quickly in the Surtsey facility. This would also be expected in the System 80+. As such, the spray mass flux becomes the important parameter governing the rate of steam condensation. The spray mass flux in the Surtsey steam condensation experiment was 0.181 kg/m^{2}s compared to 0.138 kg/m^{2}s for the System 80+ when one spray pump provides its maximum runout flow of 6,500 gpm. On the basis of this consideration, the staff believes that the test demonstrated a conservative depressurization rate. The observed rate is well below the rate the staff believes would be necessary to consider the possibility of creating detonable mixtures.

To further ensure that rapid condensation will not occur in the System 80+ design, ABB-CE in Appendix A of the System 80+ emergency operating guidelines (EOGs) state that only one containment spray train or a throttled spray train be activated when restoring sprays.

An additional consideration is the potential of generating significant concentration gradients within the containment during the course of the event. The HDR test facility experiments showed significant mixing for low-elevation release points. Stratification, however, was observed for cases with an elevated release point. Therefore, particular attention was given to high release points in the System 80+ design.

The highest release point in the System 80+ would be from the pressurizer through the pressurizer housing. Two igniters have been placed inside the top of the pressurizer housing and four more igniters are located outside the pressurizer. Therefore, the staff does not expect significant stratification within the System 80+ containment.

The HMS is designed to be manually actuated from the MCR. Actuation is expected upon recognition of an uncovered core. The presence of an uncovered core condition can be established by (1) no liquid measurement in the upper plenum, as noted by the lowest reactor vessel level monitoring system (RVLMS) sensors, (2) core exit temperature readings above 371 °C (700 °F) which are indicative of superheat, and (3) SI unavailable.

The hydrogen igniter system is designed to survive a severe accident environment. This is accomplished by locating transformers and power supplies outside of the containment and only having the igniter located within the containment. Power is supplied to igniters via cables designed for operation during a 45-minute continuous burn at 650 °C (1,200 °F).

In 10 CFR 50.34(f)(3)(v), the staff requires containment integrity to be maintained below ASME Code Service Level C limits for steel containments during an accident that releases hydrogen generated from 100-percent fuel-clad metal-water reaction. ABB-CE performed analyses, based on the methodology described in CESSAR-DC Appendix 19.11E to determine the pressurization resulting from adiabatic isochoric complete combustion of hydrogen produced by oxidizing 105 percent of the System 80+ active fuel-clad material. This was assumed to be a bounding approach. The maximum calculated containment pressure was less than 102 psia which is below the ASME Code Service Level C stress intensity of 135 psia at a temperature of 232 °C (450 °F).

The staff concludes that these analyses show that the design conforms to this regulatory requirement.

19.2.3.3.1.2 Basis for Acceptability

The System 80+ design conforms to the requirements of SECY-90-016 and 10 CFR 50.34(f)(2)(ix) by designing a system for hydrogen control that provides reasonable assurance that uniformly distributed hydrogen concentrations inside containment will not exceed 10 percent. The System 80+ design is capable of withstanding the pressurization loadings associated with the complete combustion of hydrogen produced by oxidizing 100 percent of the active fuel-clad material as required by 10 CFR 50.34(f)(3)(v).

19.2.3.3.2 Core Debris Coolability

Core debris coolability and quenchability have been the subject of extensive research over the past decade; however, much uncertainty still exists relative to this phenomenon which will most likely not be resolved in the near future. Because of this uncertainty, the NRC decided that the question is not whether coolability or quenchability has been achieved or can be achieved; but rather, what is the impact on the containment design if they are not achieved.

CCI is a severe-accident phenomenon that involves the melting and decomposition of concrete in contact with molten core debris. This phenomenon may occur following accident sequences which result in molten core debris breaching the reactor vessel and spreading onto the floor of the reactor cavity. The thickness of the layer of core debris within the reactor cavity depends upon the amount of core debris, its spreadability, and the area of the reactor cavity floor. Once on the reactor cavity floor, the molten core debris may react with the concrete and any available water producing non-condensible gases, water vapor, and heat from exothermic reactions.

CCI can challenge the containment by various mechanisms including: pressurization from non-condensible gas and steam generated, destruction of structural support members, and melt-through of the containment liner. Noncondensible gases, primarily carbon dioxide, carbon monoxide, and hydrogen, are released from the concrete as it decomposes and are formed from reactions between water and metals within the molten core debris. The core debris and concrete are heated from the combined effects of decay heat and exothermic chemical reactions.

19.2.3.3.2.1 Features to Prevent and/or Mitigate

In SECY-93-087, the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria: (1) provide reactor cavity floor space to enhance debris spreading; (2) provide a means to flood the reactor cavity to assist in the cooling process; (3) protect the containment liner and other structural members with concrete; if necessary, and (4) ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed ASME Code Service Level C limits for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from CCIs. In its July 21, 1993, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for core debris coolability is as follows:

> The standard design must include features that reduce the potential for and effect of interactions with molten core debris by:
>
> (1) providing reactor cavity floor space to enhance debris spreading;
>
> (2) providing a means to flood the reactor cavity to assist in the cooling process;
>
> (3) protecting the containment liner and other structural members with concrete, if necessary; and
>
> (4) providing design features that ensure that the best estimate environmental conditions (pressure and temperature) resulting from CCIs do not exceed ASME Code Service Level C limits for steel containments or factored load category for concrete containments, for approximately 24 hours.

ABB-CE has incorporated many features in the System 80+ design to help mitigate the effects of CCI. The following features were judged by the staff as being most important: a large reactor cavity floor area with minimal obstructions to the spreading of core debris, a CFS, an external means of adding water to the reactor cavity, use of sacrificial limestone-based concrete for the reactor cavity floor and robust reactor vessel support structures (e.g., corbels and lower cavity walls).

19.2.3.3.2.1.1 Reactor Cavity Floor Area

The System 80+ reactor cavity has been designed to maximize the unobstructed floor area available to the spreading of corium debris. The cavity floor is free from obstructions and comprises an area available for corium debris spreading of approximately 92.90 m^2 (1,000 ft^2). Approximately 64.4 m^2 (693 ft^2) of available floor area is flat while the remainder is provided by the sloped section of the core debris chamber. If the sumps are excluded, the reactor cavity has a flat floor area of 662.9 m^2 (677 ft^2). The maximum depth of core debris covering the reactor cavity floor would be less than 0.25 m (10 in.). The ratio of reactor cavity floor area to rated thermal power for the System 80+ is 0.024 m^2/MWt. This ratio is greater than the ERRI URD design criterion of 0.02 m^2/MWt for debris coolability which represents the EPRI estimate of what is required to adequately to adequately cool core debris. The staff does not support or dispute the EPRI floor sizing criteria. Instead, the staff concludes that an unobstructed floor area, along with the System 80+ design features mentioned above, provide measures to promote the potential for core debris coolability, but do not necessarily ensure it.

To determine whether the reactor cavity complies with the criteria in SECY-93-087 relative to providing reactor cavity floor space to enhance debris spreading, the staff reviewed the total floor area of the reactor cavity, the number of obstructions present to interfere with the spreading of molten core debris, and the impact of requiring further modifications to the containment design. On the basis of this evaluation, the staff finds that the design has effectively complied with the EPRI criteria by having a net floor area of approximately 0.02 m^2/MWt. In addition, ABB-CE has demonstrated that the floor area has been maximized. Further increases would require extensive redesign. Also, the floor area has minimal obstructions to interfere with the spreading of molten core debris. On the basis of these considerations, the staff concludes that the design is acceptable.

19.2.3.3.2.1.2 Cavity Flooding System

A CFS has been incorporated into the System 80+ design to supply water from the IRWST through the HVT to the reactor cavity to assist in the cooling process of core debris. The system would only be activated if corium melt-through of the reactor vessel appears to be probable. Once activated, the water is intended to flood the cavity floor before the core debris pours in. The water also cools and condenses gases evolved during CCI, thereby limiting containment temperature and pressure increases, and scrubs fission-product releases. The CFS is discussed in CESSAR-DC Sections 6.8, 19.6.3.16, and 19.11.3.3.

The CFS consists of four 30-cm (12-in.)-diameter spillways from the IRWST to the HVT and two 25-cm (10-in.)-diameter spillways that connect the HVT with the reactor cavity. The CFS valves are located approximately 1.5 m (5 ft) above the basemat to avoid direct attack from core debris. The HVT spillways and the reactor cavity spillways are equipped with remote, manually actuated MOVs that are qualified for submerged operation. The CFS is seismic Category I.

Each holdup volume flooding valve is powered from separate Class 1E channels, and each cavity flooding valve is powered from separate Class 1E divisions. The Class 1E buses are normally supplied from offsite power sources. Upon LOOP, power to the buses can be supplied by the Class 1E EDGs or the Class 1E batteries. In addition, the CTG can power these buses upon loss of all other ac power. Once the valves have been actuated, the water moves passively from the IRWST to the cavity because of the natural hydraulic driving heads of the system. Fully flooded, the water level in the reactor cavity will be approximately 5.2 m (17 ft) above the floor. The CFS has been designed to flood the reactor cavity to the 1.5-m (5-ft) level in no more than 30 minutes. The time to completely fill the reactor cavity to the equilibrium elevation was calculated to be about 72 minutes with two HVT spillway valves and one reactor cavity spillway valve open. The maximum flood level was established to avoid contact between the cavity flood water and the in-core instrument (ICI) plates below the reactor vessel lower head in case of an inadvertent actuation of the CFS.

Accident management guidance indicates that the CFS will be actuated once a potential core-melt condition is imminent or has been diagnosed as being in progress. Among indications that are capable of diagnosing core uncovery are (1) core exit thermocouple (CET) temperatures in excess of 650 °C (1200 °F), (2) RVLMS readings indicative of no liquid above the fuel alignment plate, and (3) significant changes in readings of self-powered neutron detectors.

As stated in CESSAR-DC Table 3.9-4, "Seismic I Active Valves," and 3.9-15, "Inservice Testing Safety-Related Pumps and Valves," the CFS are ASME Section III, Code Class 2 and Safety Class 2. The CFS valves are stroke tested each refueling outage in accordance with the Inservice Inspection and Testing Program, and Section XI of the ASME Code.

The staff concludes that the CFS meets the criteria of SECY-93-087 relative to providing a means to flood the reactor cavity to assist in the process of cooling core debris.

19.2.3.3.2.1.3 Containment Spray System

The CSS is a safety-grade and seismic Category I system designed to reduce containment pressure and temperature and remove iodine from the containment atmosphere following a main steamline break (MSLB), a LOCA, or a severe accident. The CSS sprays of borated water into the containment atmosphere from the upper regions of the containment. The spray flow is produced by the containment spray pumps which take suction from the IRWST. The CSS is discussed in Sections 6.5 of the CESSAR-DC and Section 6.5 of this report.

The spray headers are in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray reduces containment pressure and temperature. The CSS is designed to adequately cool the containment atmosphere to limit post-design-basis-accident building temperatures and pressures to less than the containment design values (3.7×10^2 kPa (53 psig) and 143 °C (290 °F)). The IRWST water used to spray the containment atmosphere minimizes the fission-product iodine by removing iodine by the spray droplets.

Containment spray flow can also be provided from an external source of water via a "tee" connection. Water from the containment sprays is collected in the HVT. As described in Section 19.2.3.3.2.1.2 of this chapter, the CFS is designed to direct water from the HVT to the reactor cavity to assist in the process of cooling core debris.

The staff concludes that the CSS is capable of reducing containment pressure and temperature and is another means of flooding the reactor cavity to assist in the process of cooling core debris, as specified in SECY-93-087.

19.2.3.3.2.1.4 Sacrificial Limestone-Based Concrete

Limestone concretes are calcium-carbonate based and are used in the construction of nuclear power plants. This concrete melts over a range of 1,380 °C (2,516 °F) to 1,600 °C (2,912 °F) and depending on the aggregate selection can liberate between 20 to 35 weight-percent CO_2 (bound as $CaCO_3$) and 4 to 5 weight-percent H_2O.

In CESSAR-DC Section 19.11.3.6.2, ABB-CE states that the minimum distance between the floor elevation and the embedded portion of the containment shell is 0.9 m (3.0 ft) in the reactor cavity. Directly under the reactor vessel, this distance increases to a maximum of 1.5 m (5 ft). An additional 4.6 m (15 ft) of concrete is available below the containment liner elevation. The basemat will be constructed of either limestone-common sand or limestone aggregate type concretes. Limestone-based concrete was chosen because of its superior resistance to ablation when compared to other commonly used basemat materials such as basaltic concrete. This improved ablation resistance allows ABB-CE to maintain containment integrity without further increasing basemat thicknesses. The results of the analyses provided by ABB-CE and the staff are provided in Sections 19.2.3.3.2.2.1 and 2 of this report, respectively.

The staff concludes that the 0.9-m (3.0-ft) layer of limestone-based concrete provides sufficient protection for the containment liner and that the criteria specified in SECY-93-087 relating to protecting the containment liner have been met.

19.2.3.3.2.1.5 Reactor Vessel Support Structure

The limestone-based concrete (discussed above) protects the containment liner from core-concrete attack in the axial direction. Core-concrete attack in the radial direction could affect the reactor cavity walls. The reactor cavity walls have a minimum thickness of 2.0 m (6.5 ft). Calculations performed for the System 80+ design show that the reactor vessel and the upper cavity could continue to be supported even if the entire lower cavity walls below the corbels were either eroded by corium attack or destroyed by a steam explosion. Reinforcing steel between the interface of adjacent walls with the upper reactor cavity wall will provide enough resistance through shear friction to support the reactor vessel and the upper cavity without relying on support from the lower cavity wall (see CESSAR-DC Section 19.11.3.6.2.8). The staff concludes that radial ablation of concrete is not a threat to containment integrity.

19.2.3.3.2.1.6 Containment Overpressure Protection

Paragraph (3)(iv) of 10 CFR 50.34(f) requires one or more dedicated containment penetrations, equivalent in size to a single 0.91 m (3 ft) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered containment vent system. This requirement is intended to ensure provision of a containment vent design feature with sufficient safety margin well ahead of a need that may be perceived in the future to mitigate the consequences of a severe accident situation.

ABB-CE shows that the containment is sufficiently robust to not require venting before 24 hours. However, to further improve containment performance, the System 80+ containment is equipped with two 7.6-cm (3.0-in.) diameter hydrogen purge vents which can be used to relieve containment pressure before containment pressure reaches ASME Code Service Level C. With respect to CCI, the vent could be used to prevent catastrophic overpressurization failure of the containment for severe-accident sequences involving prolonged periods of CCI. The hydrogen purge vents are capable of opening when exposed to an internal pressure corresponding to ASME Code Service Level C, of 972 kPa (141 psia) at a temperature of 177 °C (350 °F), and can be powered by the AAC source (i.e., CTG).

ABB-CE has provided this venting capability; however, they have demonstrated that venting is not needed for most of the severe-accident events. For those sequences in which venting would aid in limiting the containment pressure below ASME Code Service Level C limits, venting would not be needed before 24 hours after the onset of core damage. Therefore, the use of the hydrogen purge vent for containment pressure control is the responsibility of the COL holder's technical support center.

In Section 19.2.4 of this chapter, the staff concludes that System 80+ meets the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance. This conclusion is based on a robust containment design and design features that limit the CCFP as noted in this report. Section 19.1.3.2 of this chapter provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this chapter. In addition, by letter dated April 26, 1994, ABB-CE indicated to the NRC that a dedicated filtered containment vent is approximately seven orders of magnitude away from being cost effective as discussed in CESSAR-DC, Appendix 19A, Table 19A.2-1).

On the basis of the above considerations, an exemption in accordance with 10 CFR 50.12 exemption criteria (a)(1), (a)(2)(ii), and (a)(2)(iii) is justified.

19.2.3.3.2.2 Analyses

In SECY-93-087, the staff concluded that the evolutionary ALWRs should ensure that the best-estimate environmental conditions (pressure and temperature) resulting from CCIs do not exceed Service Level C for steel containments, for approximately 24 hours after the onset of core damage. In addition, they should ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from CCIs.

The staff judged that 24 hours was an appropriate time period to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission-product deposition in the containment through natural mechanisms, and offsite protective measures. In recognition of the uncertainties in severe-accident progression and phenomenology, that period was developed as a guideline and not as a strict criterion.

19.2.3.3.2.2.1 ABB-CE Analyses Using MAAP 3.0B

In CESSAR-DC Section 19.11.5, ABB-CE presents the results of their deter-ministic evaluation for several specific accident challenges to evaluate the containment's performance. The evaluation was performed using the Modular Accident Analysis Program 3.0B (MAAP 3), Revision 16.03, which was modified to model the configuration of the System 80+.

Using the System 80+ probabilistic risk assessment, ABB-CE chose accident sequences with relatively larger frequencies and which resulted in larger contributions to the radiological releases. These included SBO scenarios, large-break LOCAs, small-break LOCAs, total-loss-of-feedwater scenarios, SGTRs, and the ISLOCA sequence. For each accident sequence, there are several mitigating systems that could be used to prevent or reduce the release of fission products to the environment. Among these mitigating systems are battery power, CFS, containment heat removal, and containment sprays.

The results of the analyses for each accident sequence are presented in tabular and graphical form at the end of CESSAR-DC Section 19.11.5. These analyses generally indicate core debris coolability and little, if any, CCI when the CFS is actuated preceding vessel breach. CCI was predicted for accident sequences with a dry cavity. The time for failure of the containment liner due to melt-through ranged from 15 to 27 hours. However, even after liner failure, the release path is expected to be torturous because the basemat lies against the liner. On the other hand, overpressurization will result in a direct release. But the minimum time for the release from containment due to overpressurization is not expected to occur until after 50 hours. The times listed above assume that the severe-accident scenario initiates at reactor trip.

A benchmark of the containment's passive pressure capability is the loss of containment heat removal sequence analyzed by ABB-CE in CESSAR-DC Section 19.11.5.4.1.2. This analysis assumes a total loss of all ac power, including the EDGs, the CTG, and the station batteries. It was also assumed that the operator fails to actuate the CFS and that the reactor vessel fails in the presence of a dry cavity. This analysis indicated that the containment liner is breached in approximately 29 hours after reactor trip. If the operator actuates the CFS, the containment liner is not breached and fission product is released, due to containment overpressurization, 63 hours after reactor trip. If station battery power is available for 8 hours, during which time the majority of the power is directed toward maintaining auxiliary feedwater flow to the SGs, and the operator floods the reactor cavity with the CFS prior to battery depletion then an overpressure containment failure is not predicted to occur for about 80 hours.

Although the timing mentioned above began at reactor trip, the criteria mentioned in SECY-90-016 and SECY-93-087 are based on beginning with the onset of core damage. Adjusting for this change in start time would reduce the given times above by 2 to 3 hours.

19.2.3.3.2.2.2 ABB-CE Analyses Using CORCON-MOD3

ABB-CE had the Argonne National Laboratory (ANL) perform concrete erosion calculations for System 80+, using CORCON-MOD3, Version 2.26, which are described in CESSAR-DC Section 19.11.4.2.2.3.2. This code was developed to compute concrete erosion rates and profiles during severe accidents. These analyses computed heat transfer to the upper crust via mechanistic heat-transfer models which allowed for consideration of growth and depletion of the crust. These models allowed the code to select the most appropriate upper surface heat flux based on the thickness and surface temperature of the corium crust. The ANL study considered corium-concrete erosion in limestone, limestone/common sand, and basaltic concretes.

The analysis indicates that the average basemat depth will not erode by significantly more than 0.9 m (3.0 ft) in a 24-hour period following initiation of CCI, regardless of the basemat composition. Over the 24-hour interval, radial erosion was predicted to be approximately equal to axial erosion.

Deterministic calculations were also performed to find the effect of molten corium accumulating in the reactor cavity sump. The cavity sump is designed with a depth of concrete between the bottom of the sump and the containment shell of 1.0 m (3.2 ft). CORCON-MOD3 analyses of concrete erosion in a sump geometry predicted the downward erosion into concrete for a 24-hour interval to be between 0.7 m (2.3 ft) and 1.0 m (3.4 ft).

As discussed in Section 19.2.3.3.2.1.4 above, the System 80+ plant will have a 0.9-m (3.0-ft) to 1.5-m (5.0-ft) layer of limestone-based concrete above the containment liner. This concrete layer is designed to protect the containment liner from being breached in the event of significant CCI. In CESSAR-DC Section 19.11.4.2.2.3, ABB-CE provides the results of analyses which calculated the extent of radial and axial ablation using the CORCON-MOD3 code. The results are given in CESSAR DC Table 19.11.4.2.2-3 and indicate that axial ablation will not reach 0.9 m (3.0 ft) in a 24-hour period.

19.2.3.3.2.2.2 Staff Analyses Using MELCOR

The staff (through Brookhaven National Laboratory (BNL)) has performed confirmatory analysis using MELCOR to evaluate containment temperature and pressurization, radial ablation, and axial ablation.

The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the overall safety conclusions from the containment analyses.

19.2.3.3.2.2.3 Conclusions

The staff did not rely on any one specific sequence or scenario performed by ABB-CE using the MAAP 3 code nor by the staff's contractor (BNL) in determining whether the System 80+ design conformed to the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to containment overpressurization. The CFS and CSS are fundamental to prolonging the period to containment overpressurization or melt-through of the containment liner.

The staff concludes that the System 80+ design conforms to the criterion when the mitigation systems incorporated into the design, such as the CFS and CSS, are considered.

19.2.3.3.2.3 Basis for Acceptability

The System 80+ meets the criteria in SECY-93-087 and the staff's proposed applicable regulation for core debris coolability through (1) designing an unobstructed reactor cavity floor that promotes debris spreading, (2) providing a diverse and redundant means of flooding the cavity, (3) providing at least a 0.9-m (3.0-ft) layer of limestone-based concrete to protect the containment liner, and (4) providing a robust reactor cavity. Containment conditions resulting from CCI can be maintained below Service Level C for 24 hours, through incorporation of the design features listed above.

19.2.3.3.3 High-Pressure Melt Ejection

HPME and subsequent DCH is a severe-accident phenomenon that could lead to early containment failure with large radioactive releases to the environment. HPME is the ejection of core debris from the reactor vessel at a high pressure. DCH is the sudden heatup and pressurization of the containment resulting from the fragmentation and dispersal of core debris within the containment atmosphere. In addition, DCH can also lead to direct attack on the containment shell. However, direct attack on the steel shell is precluded in the System 80+ design because the steel shell is either protected by concrete or by the crane wall.

19.2.3.3.3.1 Features to Prevent and/or Mitigate

In SECY-90-016, the staff concluded that evolutionary LWR designs should have a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs should have a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with such operations as refueling, maintenance, or surveillance.

In SECY-93-087, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs should have a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

On the basis of engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris reaching the upper containment include ledges or walls that would deflect core debris and an indirect path from the reactor cavity to the upper containment. The staff position in SECY-93-087 evolved from its position in SECY-90-016 and therefore, is the basis for the staff's review and evaluation.

Therefore, the staff's proposed applicable regulation for HPME is as follows:

> The standard design must provide a reliable means to depressurize the RCS and cavity design features, to reduce the amount of ejected core debris that may reach the upper containment so that the potential for and effects of interactions with molten core ejected under high pressure are reduced.

One of the major preventive features of the System 80+ is the RDS which is discussed in CESSAR-DC Sections 6.7, 19.11.3.5, and 19.11.4.4.1.4.2. The staff's evaluation of the RDS appears in Sections 6.7 and 7.4.7 of this report. The discussions are focused on the performance of the system within the design-basis envelope. The following discussion focuses on the performance during severe-accident conditions. However, as a preface to that discussion, a brief description of the system is in order.

The RDS is a manually operated safety-grade system which will provide an initial flow capable of depressurizing the RCS from 1.72 $\times$ 10^{-4} kPa (2,500 psia) to 1.72 $\times$ 10^{-3} kPa (250 psia) in an attempt to prevent reactor vessel melt-through. This is achieved by venting steam or water from the pressurizer through two 15-cm (6.0-in.) isolation valves in each of two parallel depressurization lines to the IRWST. The RDS valves are of a size to accommodate a total loss of main feedwater and EFW event. Active RDS components are designed to be powered from a dc bus. Power connections are such that in the case of a total SBO and the loss of one battery bank, an RD bleed path from the pressurizer can be established.

Following a complete and irrecoverable total LOFW event, the operator is instructed via the EOGs to actuate the RDS valves and maintain inventory control and heat removal by means of the feed-and-bleed process. If SI is unavailable for the inventory feed operation, the operator is still instructed to perform this operation so that the SITs can inject into the RCS. Thus, the EOGs direct early actuation of the RDS in advance of core uncovery.

The RDS valves must remain open during the in-vessel phase of a severe accident to ensure that any potential vessel failure occurs at low pressure. Therefore, the RDS valves are designed to fail in the open position once actuated. Once the reactor vessel has failed, the RDS is no longer needed. ABB-CE states that the capability of the depressurization system will not be degraded by radiation exposure or thermal loads. The design-basis qualification temperature of the RDS valves is 371 °C (700 °F). ABB-CE performed analyses to determine conditions in the RCS if the operator fails to actuate the RDS. These analyses indicate that the design-basis temperature will only be exceeded for a short time preceding RCS depressurization caused by the failure of the pressurizer surge line. Because the RDS valves will be actuated before the core is damaged and because of the high likelihood that the valves will be available well into a severe accident, the staff has concluded that the RDS valves will be available to depressurize the RCS during a severe accident.

The design of the reactor cavity is expected to decrease the amount of ejected core debris that reaches the upper containment. This decrease is anticipated through (1) capture and trapping of some debris in the reactor cavity and (2) impaction and removal of core debris as it is transported between the reactor cavity and upper containment.

System 80+ is equipped with an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. The reactor cavity debris chamber and exit shaft have been designed so that following a failure of the reactor vessel, high-inertia corium debris would de-entrain and collect in the debris chamber while the lower-inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path.

One possible pathway from the reactor cavity to the upper containment would be through the instrument shaft. To minimize the possibility of corium carryover, the vertically oriented shaft has been provided with a limited gas venting area. Analyses performed by ABB, based on models developed by SNL, indicate that only 10 percent of entrained corium could be expected to initially be carried upward into the shaft. Finally, gas/corium outflow into the instrument shaft is restricted by an instrument seal table.

The entrance to the reactor cavity is a single stairway from the El. 91 ft 9 in. operating deck. This stairwell connects the upper containment with the reactor cavity by means of a convoluted pathway through an HVAC room. The staff considers this pathway sufficiently torturous to contain ejected core debris within the reactor cavity.

19.2.3.3.3.2 Basis for Acceptability

In SECY-93-087, the staff recommended that the Commission approve the general criteria that the evolutionary ALWR designs have reliable depressurization systems and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

The RDS has a reliable dc power supply to ensure its operability. The containment design is expected to decrease the amount of ejected core debris that leaves the reactor cavity. This decrease is anticipated through (1) capture and trapping of debris in the reactor cavity and (2) impaction and removal of core debris as it is transported between the reactor cavity and upper containment. On this basis, the staff concludes that the criteria of SECY-93-087 and the staff's proposed applicable regulation for HPME have been met.

19.2.3.3.4 Fuel-Coolant Interaction

The containment function can be challenged by energetic or rapid energy releases. One such energetic or rapid energy release is FCI which results in a steam explosion. The term steam explosion refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in RSG, shock waves, and possible mechanical damage. To be a significant safety concern, the interaction must be very rapid and must involve a large fraction of the core mass. Steam explosions can occur either inside (in-vessel) or outside (ex-vessel) the reactor vessel.

19.2.3.3.4.1 In-Vessel Steam Explosion

In NUREG-1116, "A Review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam Explosions," the staff summarized the deliberations of the Steam Explosion Review Group's (SERG's) understanding of the potential for containment failure arising from in-vessel steam explosions during core-melt accidents. The consensus of the SERG was that the occurrence of an in-vessel steam explosion of sufficient energetics which could lead to containment failure was of sufficiently low probability to allow it to be eliminated as a credible threat.

This conclusion was reached despite the expression of differing opinions on the modeling of basic steam explosion sequence phenomenology. An opinion supported by most members of SERG is that the probability of containment failure is reduced by the expectation of limited melt mass involvement in the explosion or low thermal-to-mechanical energy conversion, or both.

This conclusion was reaffirmed at the recent meeting of the Committee on the Safety of Nuclear Installations (CSNI), "Specialist Meeting on Fuel-Coolant," in January 1993. The conclusion of the meeting was that alpha-mode failure was highly unlikely because of the structures in the lower reactor vessel head. These structures, such as the ICI guide tubes, would limit the melt mass involvement by causing incoherent relocation of the molten corium.

19.2.3.3.4.2 Ex-Vessel Steam Explosion

In SECY-93-087, the staff stated that any dynamic forces due to ex-vessel FCI on the integrity of the containment should be evaluated. Direct containment threats due to shock pressure impulse and expansion work with possible missile generation cannot occur. The expected containment failure mode from EVSE is a steam explosion occurring within the reactor cavity that weakens and/or collapses the RV supporting structures (e.g., cavity walls, RV supports). Failure of the RV supports may lead to excessive motions in the RCS piping which can ultimately cause a containment penetration to fail.

In CESSAR-DC Section 19.11.3.6.2.8, ABB-CE performed calculations showing that the reactor vessel and the upper cavity could continue to be supported by structures adjacent to the cavity even if the entire lower cavity wall below the corbels was destroyed by a steam explosion.

This feature of the System 80+ cavity design ensures that steam explosion loadings in the reactor cavity (even those that fail the cavity lower walls) will not be sufficient to induce a failure of containment integrity.

19.2.3.3.4.2.1 Reactor Cavity Strength

ABB-CE's assessment of reactor cavity strength is documented in CESSAR-DC Section 19.11.3.6.2.7. ABB-CE calculated the impulse capacity of the concrete corbels supporting the reactor vessel and the lower portion of the reactor cavity to be 23.6 kPa-s (3.43 psi-s) and 8.96 kPa-s (1.30 psi-s), respectively. These calculations assumed a triangular forcing function and used an impulse duration of 5 milliseconds (ms), which appears to be reasonable considering pulse widths observed during FCI experiments involving corium simulates. The corbels and the cavity are stiff structures that have natural periods of 4.5 ms and 11.5 ms, respectively.

19.2.3.3.4.2.2 Staff Analysis

The staff performed an independent assessment to determine the expected range of pressure impulses resulting from steam explosions in the reactor cavity region using the TEXAS computer code. The assessment consisted of a base case and a number of parametric calculations. The assessment did not consider chemical augmentation of the steam explosion energetics. It is not currently known if such augmentation can occur with zirconium as it can with aluminum. The initial conditions and the results of the assessment are documented in the draft report ERI/NRC 94-201, "An Assessment of Ex-Vessel Fuel-Coolant-Interaction Energetics for the Combustion Engineering System 80+ Advanced Pressurized Water Reactor."

This assessment based the quantity, state, and composition of corium in the lower plenum prior to vessel breach on NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution." ABB-CE adjusted these results as shown in CESSAR-DC Table 19.11.4.1.1-1 so that they may apply to System 80+. For vessel failure at low pressure, ABB-CE estimates that 63 percent of the core relocates to the lower plenum and 45 percent of that mass is molten.

For the base case, the staff assumed a gravity pour of 35 kg/sec (77 lbm/sec) through a single 3-cm (1.2-in.)-diameter instrument tube that was artificially triggered after one second. The pressure impulses in the reactor cavity were determined to be approximately 7.0 kPa-s (1.0 psia-s) at the corbel supports and 2.9 kPa-s (0.42 psia-s) at the cavity wall. These impulses are within the capacity values quoted above.

The following parametric calculations were performed to give an idea of the possible range of pressurizations that could be expected:

- increasing the melt superheat temperature by 100 Kelvin
- increasing the number of failed penetrations from one to eight
- decreasing the pool depth from 5.5 m (18 ft) to 4.5 m (15 ft) and then to 3.0 m (9.8 ft), increasing the mass flow rate, and
- increasing the temperature of the pool to saturation

The most significant parameters appear to be the number of failed penetrations and the depth of the pool. Increasing the number of penetrations from one to eight increased the pressure impulse to 61 kPa-s (8.9 psia-s) at the corbel supports and 25 kPa-s (3.6 psia-s) at the cavity wall (these were the most limiting loads calculated). Decreasing the pool depth to 4.5 m (15 ft) slightly increased the cavity wall loading to 3.4 kPa-s (0.49 psia-s) and further decreasing the pool depth to 3 m (9.8 ft) decreased the pressure impulse seen at the cavity wall to 2.8 kPa-s (0.41 psia-s). Therefore, the impact of FCI can be diminished by reducing the depth of water in the pool.

It appears from these analyses that the System 80+ reactor cavity can accommodate the staff's best-estimate EVSE resulting from the failure of a single instrument tube and is likely to survive an EVSE involving between 1 and 8 failed instrument tubes. It appears unlikely that the reactor cavity wall directly below the corbels will survive an EVSE involving eight simultaneously failed instrument tubes and that an FCI resulting from global creep rupture of the lower vessel head will most likely result in containment failure.

19.2.3.3.4.3 Basis for Acceptability

The staff concludes that in-vessel steam explosions are not a threat to the System 80+ containment based on the conclusions reached in NUREG-1116 and reaffirmed at the recent CSNI meeting as documented in NUREG/CR-0127, "LWR Pressure Vessel Irradiation Surveillance Dosimetry Quarterly Report October - December 1977," September 1978.

To better understand the possible pressure impulses on the reactor cavity from an EVSE, the staff performed best estimate analyses with the latest analytical tools. These analyses showed that the cavity is likely to survive an EVSE involving between 1 and 8 failed penetrations. Because of the uncertainties associated with these analyses, the staff requested that ABB-CE provide further assurance that the System 80+ could accommodate an EVSE.

As discussed above in Section 19.2.3.3.4.2, structural calculations performed by ABB-CE show that the integrity of the containment will be preserved even if the reactor cavity walls beneath the corbels are destroyed. The staff believes that the ability of the containment to survive without these cavity walls is a sufficient basis to conclude that containment integrity will not be compromised by an EVSE.

19.2.3.3.5 Steam Generator Tube Ruptures

Although an SGTR is not a severe-accident phenomenon, SGTRs that proceed to core damage can contribute significantly to plant risk. A rupture of one or more SG tubes could lead to the actuation of the SG safety or relief valves, discharging primary system radioactive inventory outside the containment.

19.2.3.3.5.1 Features to Prevent and/or Mitigate

The staff concluded, in Section 15.3.9 of this report, that there is a reasonable assurance that SGTR events pose no undue threat to the public health and safety (resolution of DSER Open Item 15.3.8-1). This conclusion is based on several preventive and mitigative features of the System 80+ design (documented in CESSAR-DC Appendix 5F). Some examples of mitigative features within the System 80+ design follow: the ability to direct all the steam from the TBS to the condenser, radiation monitors, including two nitrogen-16 (N-16) monitors in the steamlines to assist in the early detection and diagnosis of SGTR events, and an RDS to limit discharge from the primary to the secondary system. The following preventive features also contributed to this safety finding: SG tubes made of thermally treated Inconel 690, which resist primary and secondary stress corrosion cracking, a reduced hot-leg temperature, a deaerator in the condensate/feedwater system for the removal of oxygen, and a condensate system with a full-flow condensate polisher to remove dissolved and suspended impurities.

A primary consideration in the staff's review was the amount of time before an MSSV lifts after the rupture of one to five SG tubes. The longer this time, the greater the time available for the operator to perform mitigative actions to prevent the lifting of an MSSV. Analyses, described in Section 4 of CESSAR-DC Appendix 5F, showed that unless the operator takes appropriate actions, MSSVs will lift after approximately four hours for a single-tube rupture. The staff determined that with the (1) modification of the component coolant water system to ensure continued operation of the TBS throughout an SGTR event, (2) addition of two N-16 monitors in the steamlines to help SGTR diagnostics, (3) addition of associated ITAACs and TS to ensure inclusion and availability of N-16 monitors, and (4) modification of emergency operations guidelines to ensure proper guidance of SGTR recovery actions, the System 80+ design has adequate diagnostics and operator response time to mitigate the consequences of SGTR events (see Section 15.3.9 of this report for additional information).

19.2.3.3.5.2 Basis for Acceptability

In SECY-93-087, the staff recommended that the Commission approve the position to require that the evolutionary PWR designs assess design features to mitigate the amount of containment bypass leakage that could result from SG tube ruptures. In its July 21, 1993, SRM, the Commission approved the staff's position. In SECY-93-087, the staff noted that the following design features were able to mitigate the releases associated with a tube rupture:

- a highly reliable (closed loop) SG shell-side heat removal system that relies on natural circulation and stored water sources

- a system which returns some of the discharge from the SG relief valve back to the primary containment

- increased pressure capacity on the SG shell side with a corresponding increase in the safety valve setpoints

ABB-CE assessed these three design alternatives in a report dated September 23, 1993, and titled, "Design Alternatives for the System 80+ Nuclear Power Plant," and found these alternatives to be cost prohibitive.

In Section 5.13, "MSSV and ADV Scrubbing," of the report on design alternatives, ABB-CE discussed the alternative of routing the discharge from the MSSVs and ADVs through a structure in which a water spray would condense the steam and remove most of the fission products, thereby reducing the consequences associated with SGTR. ABB-CE estimated a cost of $9.5 million for this system. In Section 5.27, "Venting the MSSV in Containment," ABB-CE evaluated the possibility of routing the MSSV steam releases back into the containment in order to minimize releases to the environment in SGTR events. ABB-CE judged that this alternative required a major redesign effort, posed serious design drawbacks, and was prohibitively expensive. In Section 5.25, "Increase Secondary Side Pressure," ABB-CE investigated the possibility of increasing the design pressure of the secondary system, including the MSSVs from 8.274×10^{-3} kPa (1,200 psia) to 1.034×10^{-4} kPa (1500 psia) in order to reduce the frequency of SGTR events. ABB-CE judged that this alternative posed serious design drawbacks with limited benefits and was prohibitively expensive.

In Section 19.4.2.1 of this chapter, the staff concluded that as a result of the low estimated CDF and associated risk levels for the System 80+, any potential modifications that cost more than about $20,000 would not be cost effective, even if the design modification were to totally eliminate the severe accidents or their consequences. Therefore, it is the staff's position that these design alternatives are impractical and would have excessive impact on the plant.

On the basis of the preventive and mitigative features in the System 80+ design, the staff concluded that the CDF has been reduced from 1×10^{-5}/year for the System 80 to 3×10^{-7}/year for the System 80+. The staff further concluded, in Section 15.3 of this report, that as a result of these preventive and mitigative features with the System 80+ design modifications and operational requirements discussed in Sections 19.2.3.3.5.1 and 15.3.9, there is reasonable assurance that SGTR events pose no undue threat to the public health and safety, and the System 80+ design satisfies the staff's proposed applicable regulation for SGTRs (see Section 15.3.9). The staff further concludes that the three design alternatives identified in SECY-93-087 have been adequately assessed and that the criteria of SECY-93-087 have been adhered to.

19.2.3.3.6 Equipment Survivability

In SECY-93-087, the NRC requires that, during the review of the credible severe-accident scenarios for ALWRs, the staff evaluate the ALWR design certification applicant's identification of the equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. ABB-CE addresses equipment survivability in CESSAR-DC Section 19.11.4.4.

Design bases events are defined as conditions of normal operation, including AOOs, DBAs, external events, and natural phenomena for which the plant must be designed. Safety-related equipment, both electrical and mechanical, must perform its safety function during design bases events. CESSAR-DC Section 3.11 defines the environmental conditions with respect to limiting design conditions for all safety-related mechanical and electrical equipment. The common terminology used for the level of assurance provided for equipment necessary for design bases events is "environmental qualification" or "equipment qualification."

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents. The environmental conditions resulting from these events are generally more limiting than those from design bases events. The NRC established a criterion to provide a reasonable level of confidence that the necessary equipment will function in the severe accident environment for the time span for which it is needed. This criterion is commonly referred to as "equipment survivability" and is fundamentally different from equipment qualification.

The applicable criteria for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents are provided in 10 CFR 50.34(f).

- Part 50.34(f)(2)(ix)(c) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

- Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

- Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

- Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment Survivability section of SECY-90-016. In its SRM of June 26, 1990, relating to SECY-90-016, the Commission approved the staff's position that features provided only (not required for DBAS) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be DBAs in the traditional sense that DBAs have been treated in the past.

However, mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function. Therefore, the staff's proposed applicable regulation for equipment survivability is as follows:

> The standard design must include analyses, based on best available methods, to demonstrate that:
>
> Equipment, both electrical and mechanical, needed to prevent and mitigate the consequences of severe accidents is capable of performing its function for

the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function.

Instrumentation needed to monitor plant conditions during a severe accident is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the instrumentation is relied upon to function.

According to SECY-90-016, ABB-CE was to review the various severe accident scenarios analyzed and identify the equipment needed to perform various functions during a severe accident and the environmental conditions under which the equipment must function. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., SBO and earthquakes) and the environment (e.g., pressure, temperature and radiation) in which the equipment is relied upon to function.

19.2.3.3.6.1 Severe Accident Environmental Conditions

In SECY-90-016, the NRC stated that mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. To address equipment survivability in environmental conditions associated with in-vessel and ex-vessel severe accidents, ABB-CE identified three severe-accident environmental conditions for which mitigative features and instrumentation, necessary to monitor the course and mitigate the consequences of the severe-accident, must survive. The first environment is a core damage accident with the core retained in-vessel, attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction, in accordance with 10 CFR 50.34(f)(2)(ix)(c). The second environment is a core melt with vessel failure created by a large-break LOCA with a wet cavity. The third expected environment is that associated with a hydrogen burn.

ABB-CE defined the in-vessel severe accident environment using MAAP 3 analyses which were supplemented with more conservative hand calculations. Using MAAP 3, ABB-CE calculated a containment temperature of 110 °C (230 °F) when 50 percent of the fuel-clad metal-water reaction has taken place. At this point, the computer model is unable to calculate further fuel-clad metal-water reaction because of a lack of water. ABB-CE extrapolated MAAP 3 analyses to assess the in-vessel gas and containment conditions should the oxidation process involve 100 percent of the active cladding. ABB-CE found the gas temperature in the RV upper plenum to be less than 1371 °C (2500 °F) and the maximum temperature inside containment to be 168 °C (335 °F) if 100 percent of the active cladding were oxidized. ABB-CE calculated the gas temperature in the area above the upper guide structure support plate (which is above the RV upper plenum and houses the RCS temperature and level monitors) to be less than 871 °C (1600 °F). This calculated profile was selected as the environmental condition for an in-vessel event.

ABB-CE also used MAAP 3 to determine the ex-vessel environmental conditions. Of the sequences analyzed with MAAP 3, the highest temperatures and pressures were for the large break LOCA with a wet cavity sequence. No credit was taken for containment sprays, but the CFS was assumed to be operable. For this sequence by use of MAAP 3, ABB-CE calculated a temperature of 166 °C (330 °F) at approximately 24 hours which subsequently rose to 177 °C (350 °F) with a pressure corresponding to ASME Code Service Level C (972 kPa (141 psia) at 177 °C (350 °F)) at approximately 48 hours after reactor scram. This calculated profile was selected as the environmental condition for an ex-vessel event.

ABB-CE established two separate hydrogen burn environments. The local burn environment was based on analyses using the MAAP 4 containment model. These results were then confirmed by comparing them to observations made during the Hydrogen Control Owners Group 1:4 scale Mark III igniter system experiment. As a result of this exercise, the following trends were identified: (1) containment temperatures away from the hydrogen source regions were below 166 °C (330 °F), (2) burning occurred at low hydrogen concentrations with long-term hydrogen concentrations controlled near 5 volume percent, and (3) containment pressures during igniter operation could be maintained below 34 kPa (50 psia). ABB-CE concluded that the mitigative features and instrumentation listed in CESSAR-DC Tables 19.11.4-1 and 2 should be located a minimum of 3.0 m (10 ft) from an igniter in order to protect this equipment from the effects of a local burn.

The second hydrogen environment assessed by ABB-CE was a global burn. In this calculation, ABB-CE assumed the following: the containment is steam inerted, 100 percent of the active cladding has been oxidized, containment sprays are recovered and the containment is de-inerted, the mixture burns at the minimum flammability point, and the combustion completeness of the mixture is 50 percent based on hydrogen and steam concentrations. On the basis of these assumptions, ABB-CE concluded, the limiting global hydrogen burn can be approximated by a temperature of 121 °C (250 °F) rising to 316 °C (600 °F) over 30 seconds and rapidly decaying over the next 10 seconds to 121 °C (250 °F). This profile was used as the environmental condition associated with a global burn of hydrogen generated by the equivalent of a 100 percent metal water reaction.

The staff had several severe accident sequences analyzed with its computer model, MELCOR, to confirm the ability of the computer model used by ABB-CE, MAAP 3, to predict the environmental conditions attendant with a severe accident. On the basis of this confirmation, the staff concludes that the environmental conditions predicted above by MAAP are acceptable approximations of the environmental conditions for which mitigative features and instrumentation, identified in this section, must survive.

Severe Accident Source Term for System 80+

Post-severe accident radiation environments can be defined as a condition resulting from accidents involving substantial core damage, including total core melt, slump and vessel bottom head failure. Two distinct classes of accidents are possible. The first includes the accidents that are "recoverable," i.e., the core degradation process could be successfully arrested and the bottom head failure would have not occurred. The other is a class of "non-recoverable" accidents that includes the scenarios, where the vessel failure could not have been prevented, and the core debris would be relocated to the reactor cavity leading to the interaction with concrete or existing water, or both.

The "recoverable" accidents result in fission product releases to the containment from three sources: (i) coolant activity, (ii) gap release, and (iii) early in-vessel release. As a source term for these releases, the applicant used the ones provided in draft NUREG-1465. That makes the "recoverable" environment identical to that of DBA Level 2, as described in CESSAR-DC Section 3.11. As a result, ABB-CE's position is that the equipment qualification done for the DBA Level 2 condition envelops that for the "recoverable" environment.

The "non-recoverable" accidents result in fission product releases to the containment from all five phases of a severe accident as described in draft NUREG-1465, i.e., (i) coolant activity, (ii) gap releases, (iii) early in-vessel release, (iv) release for the interaction between core debris and concrete, and (v) late in-vessel release.

The equipment and instrumentation can be exposed to the radiation field from two sources, i.e., containment atmosphere and water. The fission products released into the atmosphere are subjected to the removal by containment spray. For the severe accident condition, ABB-CE uses the best estimate spray removal coefficients, as described in CESSAR-DC Sections 15.6.5.4 and 15.6.5.5. The fission product released during the CCI would be scrubbed by the overlaying pool of water. ABB-CE applies the decontamination factor (DF) of 10 for the pool. The fraction of the fission product that were not scrubbed by the overlaying pool are released to the containment atmosphere. For the water source doses, the fission product released from all five phases is deposited directly in the IRWST water.

The equipment and instrumentation required for severe accident mitigation and monitoring, thus exposed to the maximum radiation field, are listed in CESSAR-DC Table 19.11.4.4-6. The instrumentation is qualified at the maximum dose for 180 days, which is 80 days longer than a period of time widely excepted as sufficient for qualification under any circumstances. The equipment will be qualified at maximum dose for 10 days. The rational for this is, that under the worst case condition, the containment may lose its integrity within 1 to 3 days. If this worst case accident progression cannot be controlled within this period of time, the operability of any equipment may not be needed. Therefore, ABB-CE postulates, the qualification for 10 days at the maximum dose is sufficient.

The staff finds the above approach acceptable based on the following:

(1) the five elements considered by the applicant bound all possible accidents including the most limiting "non-recoverable" events (this concept is endorsed by draft NUREG-1465),

(2) the "recoverable" accident means, that the in-vessel core degradation was arrested by flooding the reactor vessel with water, therefore, only the first three, out of five, are applicable; (i) there is no ex-vessel source possible, and (ii) the late in-vessel releases are impossible because of the lower system temperature associated with a recoverable event,

(3) direct deposition of the fission product in the IRWST maximizes the concentration of radioactive materials, thereby maximizing the calculated dose for the equipment and/or instrumentation exposed to the water source activity,

(4) similarly, the radioactive source within the containment atmosphere is maximized by assuming minimal removal rate of the aerosol,

(5) the decontamination factor (DF) of 10, applied to the fission product scrubbing by the overlying pool, is consistent with SRP Section 6.5.5,

(6) independent calculations, performed by the staff with the MELCOR computer program, support ABB-CE's claim that the loss of containment integrity can happen within 3 days of a postulated accident. After the loss of containment integrity, any mitigative feature becomes basically ineffective. It is, therefore, the staff's position that to account for existing uncertainties, a period of 10 days is sufficient for the qualification at the maximum dose.

19.2.3.3.6.2 Equipment and Instrumentation Necessary to Survive

ABB-CE identified the instrumentation and equipment required for severe accident mitigation and recovery in CESSAR-DC Tables 19.11.4.4-1 and 2. The equipment listed is necessary to ensure that adequate inventory and heat removal can be provided to the RCS, reactivity control can be maintained, hydrogen can be controlled, and containment heat removal via sprays is functional. The list of equipment also includes the CFS and the containment penetrations in the case of an ex-vessel event. The instrumentation was chosen so that the operator could confirm and trend the results of actions taken and that adequate information would be available for those responsible for making decisions about accident management.

The instrumentation and equipment required for severe accident mitigation and recovery will be demonstrated to operate in the applicable environment described above in Section 19.2.3.3.6.1. The demonstration process used to provide reasonable assurance that the instrumentation and equipment will operate includes one or more of the following factors: limited time period in or exposure to the environment, the use of similar equipment in commercial industry exposed to a similar environment, the use of analytical extrapolations, the use of vendor performance data, the use of procurement specifications imposed on the vendor, or the results of tests performed in the nuclear industry or at independent laboratories.

Two exceptions to this requirement are the high range radiation monitor and the containment temperature monitor which are qualified to 166 °C (330 °F) and 690 kPa (100 psia) for 24 hours per 10 CFR 50.49 environmental qualification requirements. Radiation inside the containment is monitored beyond 24 hours by direct containment air sampling using the post accident sampling system (PASS). The PASS, the majority of which is located outside containment, is qualified to 177 °C (350 °F) and 972 kPa (141 psia) for 48 hours. Containment temperature was not seen as a significant enough parameter to require qualification beyond 10 CFR 50.49 because the most likely accident management decision to be made after 24 hours would be to vent the containment. The parameters necessary to make this decision are containment pressure and radiation which would be available. Hydrogen purge valves are capable of opening under a pressure, corresponding to ASME Code Service Level C, of 972 kPa (141 psia) and a temperature of 177 °C (350 °F).

The staff performed an independent assessment of the entire list of equipment and instrumentation in CESSAR-DC Tables 19.11.4.4-1 and 2 and compared them to the more extensive lists such as that required by RG 1.97 and 10 CFR 50.34(f) to ensure that the equipment and instrumentation provided is sufficient. The staff concludes that the equipment and instrumentation needed to perform and monitor the mitigative functions necessary during a severe accident are adequate.

19.2.3.3.6.3 Basis for Acceptability

In SECY-93-087, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs review the various severe accident scenarios analyzed and identify the equipment needed to perform its function during a severe accident and the environmental conditions under which the equipment must function. In its July 21, 1993, SRM, the Commission approved the staff's position.

On the basis of the equipment and instrumentation identified in CESSAR-DC Tables 19.11.4.4-1 and 2 and the commitment that reasonable assurance will be provided that the equipment and instrumentation will operate in the applicable environments listed in Section 19.2.3.3.6.1 of this chapter, the staff concludes that the criteria of SECY-93-087 have been complied with and the requirements of 10 CFR 50.34(f) discussed in Section 19.2.3.3.6, and the staff's proposed applicable regulation for equipment survivability have been satisfied.

19.2.4 Containment Performance

The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the CPG. The CPG ensures that the containment would perform its function in the face of most severe-accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident.

Two alternative CPGs were identified in SECY-90-016: a CCFP of 0.1 or a deterministic CPG that offers comparable protection. In its June 26, 1990, SRM, the Commission approved the use of the 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary ALWRs. In assessing the probability of containment failure, two definitions of containment failure were considered. These include a CCFP based on structural integrity and on a dose definition. Using the dose definition of containment failure, the CCFP for the System 80+ is approximately 3 percent while using the shell integrity definition of containment failure results in a CCFP of 11 percent, which is slightly higher than the goal. In Section 19.1.3.2.1 of this chapter, the staff discusses the results of the CCFP analyses. Through these analyses, the staff concludes that because of the approximate nature of the CPG, the recognition that PRA results, particularly bottom-line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11-percent CCFP estimate are late, containment basemat melt-throughs rather than releases to the atmosphere, the System 80+ design limits the CCFP to approximately 0.1.

The Commission directed that the use of a 0.1 CCFP should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. Therefore, the staff's proposed applicable regulation for containment performance is as follows:

> The standard design must include design features to limit the CCFP for the more likely severe accident challenges.

Section 19.1.3.2 of this chapter provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this chapter. Based on the evaluations in these sections, the staff concludes that the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance have been met.

19.2.5 Accident Management

Accident management (AM) encompasses those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases. AM, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel-damage regimes, and by making full use of existing plant equipment and operator skills and creativity to terminate severe accidents and limit offsite releases.

On the basis of PRAs and severe-accident analyses for the current generation of operating plants, the NRC staff concluded that the risk associated with severe accidents could be further reduced through improvements to utility accident management capabilities. Although future reactor designs such as System 80+ will have enhanced capabilities for preventing and mitigating severe accidents, accident management will remain an important element of defense-in-depth for these designs. However, the increased attention to accident prevention and mitigation in these designs can be expected to alter the scope, focus, and overall importance of accident management relative to that for operating reactors. For example, increased attention to accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention, while increasing the time available for such action if necessary. This will tend to relieve operators of the need for making rapid decisions, and will permit a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), human intervention and accident management will continue to be needed.

In SECY-88-147 and Generic Letter (GL) 88-20, the staff identified the development of an "accident management plan" by each operating reactor licensee as a key element of severe-accident closure. The major goals, framework, and elements of an accident management plan was subsequently described in SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs," and in an NRC letter to Nuclear Management and Resources Council dated July 29, 1991. The AM plan provides a framework within the licensee's organization for evaluating information on severe-accidents, for preparing and implementing severe accident operating procedures, and for training operators and managers in these procedures.

The nuclear power industry initiated a coordinated program on accident management in 1990. As described in SECY-90-313, "Status of Accident Management Program and Plans for Implementation," this program involves the development of three major products: (1) a structured method by which utilities may systematically evaluate and enhance their abilities to deal with potential severe accidents, (2) a technical-basis document that distills the results of earlier technical studies related to accident management and summarizes applicable technology and results, and (3) vendor-specific accident management guidelines for use by individual utilities in establishing plant-specific accident management procedures and guidance. The program was subsequently broadened to include the development of guidance and material to support utility activities related to training in severe accidents. As described in SECY-93-308, "Status of Implementation Plan for Closure of Severe Accident Issues," the industry accident management program is scheduled for completion in 1994. Using the guidance developed through this program, a plant-specific accident management plan is expected to be implemented at each operating plant by 1997 as part of a binding industry initiative.

For both operating and advanced reactors, the overall responsibility for AM, including development, implementation, and maintenance of the accident management plan, lies with the nuclear utility, since the utility is ultimately responsible for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations. However, the development and implementation of accident management in future reactors involves both the reactor designer and the plant owner/operator, particularly in view of the fact that many of the design details are still to be developed (such as balance of plant equipment and final piping layout). The plant designer is responsible for developing the technical bases for the plant-specific accident management program or plan, whereas the owner/operator is responsible for developing and implementing the complete accident management plan, including those areas beyond the purview of the plant designer, such as the content and techniques for severe-accident training, and the delineation of decisionmaking responsibilities at a plant-specific level.

The COL applicant should develop and submit an accident management plan as part of the COL application. The plan should include ABB-CE's commitments to perform a systematic evaluation of the plant's ability to deal with potential severe accidents, and to implement the necessary enhancements within the detailed plant design and organization, including severe-accident management guidelines and training. General areas that should be addressed in the plan are (1) accident management strategies and implementing procedures, (2) training in severe accidents, (3) guidance and computational tools for technical support, (4) instrumentation, and (5) decisionmaking responsibilities.

All System 80+ PRA insights and COL action items that fall within the scope of accident management should be specifically addressed as part of the COL applicant's accident management plan, including the following:

- development of detailed guidance and procedures for the use of the severe-accident design features in the System 80+ design, including the RDS, the HMS, the reactor cavity flood system, the ECSBS, and the hydrogen purge vent

- development of additional guidance and procedures on protection of fission-product barriers, including the following:

  - filling of the SGs to prevent a thermally induced SGTR

  - depressurization of a SG to effect early closure of a cycling MSSV following a SGTR with core damage

  - use of spray systems for containment fission-product scrubbing

  - use of the annulus ventilation and filtering system to control fission-product releases following intact as well as vented severe-accident sequences

- evaluation of (1) information needed to implement the accident management guidelines and (2) plant instrumentation that could be used to supply the needed information, considering availability and survivability under severe-accident conditions

The staff will review the accident management plan at the COL stage to ensure that the evaluation process and commitments proposed by the COL applicant provide an acceptable means of systematically assessing, enhancing, and maintaining AM capabilities, consistent with staff expectations.

The COL applicant would subsequently implement the plan and submit the results for staff review before plant operation. This plan should be developed according to the final as-built plant, the accident management-related information developed by the plant designer, and the accident management program guidance developed for the current generation of operating reactors. This is COL Action Item 19-15.

19.2.6 Containment ASME Service Level C and Ultimate Pressure Capability

19.2.6.1 Introduction

In CESSAR-DC Section 19.11, ABB-CE discusses the severe-accident phenomenology and the containment performance under postulated severe-accident conditions for the System 80+. The staff presents its evaluation of the adequacy of the containment to withstand the postulated design-basis loads in Sections 3.8.1 and 3.8.2 of this report. The evaluation that follows covers the structural performance aspect of the steel containment under severe-accident conditions. Specifically, the purpose of this evaluation is to assess (1) the performance of the containment under severe-accident conditions against the containment performance criteria in the Commission paper, SECY-90-016, (2) the potential local leakage, and (3) the adequacy and acceptability of the proposed fragility curve for the System 80+ containment.

In SECY-90-016, the staff recommended that the Commission approve a CCFP of 0.1 given the occurrence of a core-melt accident, or a deterministic CPG that offers comparable protection in the evaluation of evolutionary ALWRs. The staff recommended the following general criterion for containment performance during a severe-accident challenge for evolutionary ALWRs in place of a CCFP of 0.1:

> The containment should maintain its role as a reliable leak-tight barrier by ensuring that containment stresses do not exceed ASME Code Service Level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24-hour period, the containment should continue to serve as a barrier against the uncontrolled release of fission products.

In its June 26, 1990, SRM, the Commission approved the use of the 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary ALWRs. The Commission further encouraged the staff to develop suitable alternative, deterministically established, containment performance objectives that provide comparable mitigation capability, if these are submitted by applicants.

The staff reviewed the containment performance evaluations submitted through the CESSAR-DC, ABB-CE's responses to NRC staff RAIs, and other submittals, which include the materials presented and discussed during the NRC meetings with ABB-CE. The staff evaluated the containment internal pressure resisting capacities corresponding to the deterministic criteria in SECY-90-016 (i.e., Service Level C limit for the first 24 hours and prevention of containment rupture or collapse to ensure that no uncontrolled release of radioactive materials after 24 hours from onset of core-melt accident) and the acceptability of the containment fragility curve submitted by ABB-CE. The details of the staff's evaluation based on the documents reviewed and the staff's understanding of the discussions in meetings are discussed below.

19.2.6.2 Deterministic Evaluation of Containment Capacity

The System 80+ severe-accident mitigation design features include (1) a large, dry, steel primary containment, (2) a reinforced-concrete secondary containment with an annuls ventilation filtration, (3) a reactor CFS, (4) a hydrogen-mitigation system to prevent in-containment hydrogen concentration from reaching detonation levels, (5) an SDS, (6) a large reactor cavity designed for retention and cooling of core debris, (7) missile-protection structures, and (8) an integrated SDC and CSS. The objective of this section is to assess the extent to which the System 80+ steel containment vessel (SCV) meets the deterministic CPGs of SECY-90-016.

(1) <u>Description of Containment</u>

The System 80+ steel containment is a spherical, welded, steel shell structure designed in accordance with Section III, Subsection NE of Division I of the ASME Code. The containment is 60.96 m (200 ft) in diameter and is constructed of steel plate with a nominal thickness of 4.445 cm (1.75 in.). The plate thickness of 5.08 cm (2 in.) is used for the anchorage region.

The material of construction is SA537 Class 2 carbon steel. Above El. 28 m (91 ft 9 in.), the containment is designed as an independent, free-standing structure. Below this elevation, the vessel is encased between the base slab of the internal structures and the shield building foundation. Shear bars are welded to the containment vessel in the embedded region to restrain it from sliding. Near the top of the embedment in concrete there is a transition region outside the shell which is filled with compressible material.

(2) <u>Deterministic CPGs Under Severe-Accident Conditions</u>

(a) For the first 24 hours after the onset of the core-melt accident, the ASME Code Section III, Division I, Subsection NE, Service Level C limit stress intensities should not be exceeded.

(b) After the first 24-hour period, the ultimate containment capacity analysis will be used to demonstrate that the containment will neither rupture nor collapse under the prevailing accident environment which could lead to uncontrolled release of radioactivity.

(3) <u>Containment Pressure Capacity Analysis</u>

(a) <u>Design-Basis Pressure Capacity</u>

For the determination of the required containment shell thickness under the design-basis pressure, the stresses in the containment for the load combinations of SRP Section 3.8.2.II.3 should be shown to satisfy the limits prescribed in SRP Table 3.8.2-1. ABB-CE has performed a 3-D finite element analysis of the containment for the load combinations identified above. The compressible material in the transition region was determined to have a modulus of 488.6 kPa/cm (180 psi/in.). The 3-D finite element model (FEM) included the equipment hatch and personnel air locks. The analysis results show a containment pressure capacity of 466.77 kPa (53 psig).

For the consideration of containment shell under compression, SRP Section 3.8.2 provides guidelines that the stresses in the containment should satisfy the stability requirements of ASME Code Section III, Subsection NE, and RG 1.57 for load combinations resulting in compressive stresses in the shell. ABB-CE performed a buckling analysis of the containment for the compressive loading combination. This analysis was performed on a 3-D model with the ANSYS finite element code using a large deflection option. The loading applied included the combined gravity load, an external pressure due to an inadvertent actuation of containment spray, and SSE loads. This buckling analysis led to an external differential pressure capacity of 13.79 kPa (2 psid) in combination with the SSE and gravity loads.

On the basis of the analyses described above, the System 80+ design- basis pressure limit for containment internal pressurization was determined to be 466.77 kPa (53 psig), and the differential pressure limit for containment external pressurization was calculated to be 13.79 kPa (2 psid) for LOCA load combination with the SSE and normal operating condition.

(b) <u>ASME Code Service Level C Limit Stress Evaluation</u>

To demonstrate compliance with the first part of the deterministic criterion of SECY-90-016, ABB-CE performed an evaluation to determine the containment pressure that may be reached without exceeding the ASME Code Service Level C limit allowable stress intensities.

The calculations based on the results of the 3-D ANSYS model with major penetrations reflected in the model, that is, the design-basis pressure capacity analyses, indicate that the pressure capacities corresponding to the ASME Code Service Level C limit criteria range from about 999.74 kPa (145 psia) at an average steel shell temperature of 143.3 °C (290 °F) to 930.79 kPa (135 psia) at a temperature of 232.2 °C (450 °F). The staff performed an independent calculation for the pressure limits based on the primary membrane stress and it showed pressure capacities of 1145.43 kPa (166.1 psia) at 143.3 °C (290 °F) and 1082.68 kPa (157 psia) at 232.2 °C (450 °F). The staff's pressure capacity assessment is based on the maximum membrane stress value and does not incorporate detailed evaluation of local higher stresses at points of change of geometry and discontinuities. Pressure capacity comparisons for the ASME Code Service Level C limit (i.e., $P_{applicant}$ = 999.74 kPa (145 psia) vs. P_{staff} = 1145.43 kPa (166.1 psia) at 143.3 °C (290 °F) and $P_{applicant}$ = 930.79 kPa (135 psia) vs. P_{staff} = 1082.68 kPa (157 psia) at 232.2 °C (450 °F)) show that ABB-CE's calculations are more detailed and conservative and are, therefore, acceptable.

Since the pressures computed above envelope the severe-accident pressure and temperature calculated by ABB-CE using the MAAP 3.0B for the first 24 hours (i.e., the peak pressure and temperature within the first 24 hours of accident are reported to be 634.41 kPa (92 psia) and 226.7 °C (440 °F)), the first 24-hour goal for containment performance under severe accidents is met. The staff believes that compliance with the deterministic criteria of SECY-90-016 will provide reasonable assurance that the containment will withstand the effects of a severe accident occurring within the containment without impairment of its structural integrity or loss of the required safety functions.

(c) <u>Deterministic Ultimate Capacity Analysis</u>

The deterministic ultimate strength of the containment shell was established by ABB-CE using an axisymmetric (2-D) shell model with added local masses to represent the shell penetrations. The stress-strain curve for material with chemistry similar to that of SA537 Class 2 indicates a relatively flat plateau of yield stress of 11.79 MPa (81.3 ksi) for strains ranging from 0.002 to 0.006. This is followed by the strain-hardening up to a maximum stress of 13.71 MPa (94.5 ksi) at a strain of 0.079. The first portion of the strain-hardening is approximately linear, with a stress level of 13.05 MPa (90 ksi) at a strain of about 4 percent. ABB-CE assumed that the ultimate containment failure occurs once the global shell stress exceeds the yield point where the strain changes from 0.002 to 0.006 without appreciable pressure increase.

As a result of this flat stress-strain curve, once the yield point of the material is exceeded, the shell may rapidly grow by up to 7.2 in. ABB-CE considered this growth to be sufficient to create a small area of separation or openings between the shell and some of the larger penetrations, thereby, leading to potentially uncontrolled release of radioactivity. Therefore, ABB-CE's assumption is conservative. ABB-CE showed an ultimate capacity of 1082.47 kPa (157 psia) at 143.3 °C (290 °F) and 1013.52 kPa (147 psia) at 232.2 °C (450 °F). The staff considers this reasonable and acceptable.

19.2.6.3 Evaluation of Containment Ultimate Capacity via Use of Fragility Curve

ABB-CE also assessed the containment median ultimate capacity via development of a containment fragility curve. The staff considers the median fragility of the containment structure as an adequate criterion for satisfying the second part of the SECY-90-016 objective, as long as the total leakage from penetrations and other bypasses are reasonably controlled.

Median Ultimate Capacity Evaluation

(1) Structural Analysis Methodology

The median ultimate containment strength was established by pressurizing the axisymmetric shell model until the median material yield stress was reached. The median ultimate containment strength was conservatively established at 10 percent above the minimum material yield strength based on 122 steel coupon test data. This assumption is judged to be realistic since it is based on the actual test data, therefore, it is acceptable. From the 2-D analysis discussed above at a temperature of 143.3 °C (290 °F), the minimum yield pressure is computed as 1082.4 kPa (157 psia) and, accordingly, the median failure pressure is determined as 1201.1 kPa (172 psia). Median ultimate containment failure pressures for a range of containment temperatures are tabulated as follows:

Temperature	Yield Stress	Median Failure Pressure
65.5 °C (150 °F)	436.1 MPa (63,250 psi)	1296.2 kPa (188 psia)
143.3 °C (290 °F)	298.0 MPa (57,728 psi)	1201.1 kPa (172 psia)
176.7 °C (350 °F)	387.5 MPa (56,210 psi)	1158.3 kPa (168 psia)
232.2 °C (450 °F)	370.1 MPa (53,680 psi)	1103.1 kPa (160 psia)

These median ultimate capacities are considered conservative and are, therefore, acceptable.

(2) Construction of Containment Fragility Curve

In order to develop a fragility curve, ABB-CE assumed that the containment failure would follow, once material yield strength is exceeded on a global basis. On the basis of 122 test data of the containment steel material, SA537 Class 2 steel, the mean yield strength is 476.43 MPa (69.1 ksi) with a standard deviation of 22.75 MPa (3.3 ksi). As discussed before, the mean yield value of the shell material needed for the assessment of the median ultimate pressure analysis was set at 110 percent of the ASME Code minimum yield strength of 413.69 MPa (60 ksi) for the SA537 Class 2 carbon steel plate. Additionally, the maximum yield strength was set by ABB-CE at 120 percent of the minimum value.

The containment fragility estimate was derived from analyses for the design-basis loads and load combinations and the severe-accident load combination in conjunction with the use of the conservative material property assumptions made above. ABB-CE conservatively assumed that the containment failure probabilities are (1) 0 percent for pressures up to 1.5 times design-basis pressure based on NUREG-1150, Volume 2, (2) 3 percent for pressure corresponding to the ASME Code Service Level C limit stress allowable obtained from the 3-D analysis, (3) 5 percent for pressure associated with the ASME Code Level C Service Limit stress as obtained from the 2-D analysis, (4) 50 percent for pressure corresponding to the nominal yield stress (using 1.1 times ASME Code minimum yield strength), and (5) 100 percent at the pressure corresponding to the maximum yield stress (using 1.2 times ASME Code minimum yield strength). The acceptability of the failure probabilities proposed above is discussed next.

The 50-percent failure probability is acceptable because a conservative mean yield strength is used based on the actual test data. The chosen vs. actual values are $1.1\sigma_{y,\,min}$ for mean chosen yield strength vs. $1.15\sigma_{y,\,min}$ for actual test value. The 100-percent failure probability is acceptable because it is also determined from a conservatively selected mean yield strength (i.e., $1.2\sigma_{y,\,min}$ (selected value) vs. $1.33\sigma_{y,\,min}$ (test-obtained value)). The acceptability of the other failure probabilities is discussed below.

ABB-CE used a linear fit between the points discussed above for the definition of the fragility curve used in the PRA. ABB-CE also compared the fragility curve above with a lognormal distributed one anchored to a best estimate of the containment pressure capacity, and concluded that the linearly fitted fragility curve overestimated the failure probability of the containment in the tail region of the fragility curve below the 3-percent failure point (in the pressure region between 648.10 kPa (94 psia) and 999.74 kPa (145 psia)).

For the fragility curve with a lognormal probability distribution, ABB-CE estimated the uncertainties associated with the median fragility value using engineering judgment and the results from an earlier analysis. The uncertainties in the prediction of the failure pressure generally arise from uncertainties in modeling and material strength. The lognormal distribution is selected to characterize the fragility curve and is defined in terms of the median pressure capacity and the combined logarithmic standard deviation. The logarithmic standard deviations from uncertainties for modeling and material properties of steel structures (β_m and β_s) are estimated as 0.10 and 0.09, respectively, by ABB-CE, and the combined logarithmic standard deviation (β_c) is estimated to be 0.137.

In NUREG/CR-2442 "Reliability Analysis of Steel Containment Strength", June 1982, the resistance modeling error is taken as $X_o = \Delta\delta$ where δ represents the basic variability of the theoretical resistance model with respect to experimental results and Δ represents the variability between experimental results and in-service condition. The total coefficient of variation (COV) [defines as $(\exp(\beta^2)-1)^{1/2}$] of resistance modeling error X_o is $V_o^2 = V_\Delta^2 + V_\delta^2$. ABB-CE used V_Δ and V_δ as 0.09 and 0.05, respectively, which would result in a $V_o^2 = 0.0106$. Therefore, the use of 0.10 for β_m is judged to be acceptable because it is consistent with NUREG/CR-2442.

The use of 0.09 for β_s is acceptable because the variability associated with material strength is expected to be the same regardless of the material temperature, and the statistical data for SA-537, Class 2 show that the average yield strength of the material is 69.1 ksi and the standard deviation is 3.3 ksi. The COV is 0.048, which is less than 0.09 used for β_s.

Therefore, the use of the combined logarithmic standard deviation (β_c) of 0.137 ($(\beta_m^2 + \beta_s^2)^{1/2}$) is acceptable.

On the basis of the preceding discussion, the staff considers the median fragility value proposed for the SCV of 1244.5 kPa (180.5 psia) (based on the actual test data) at 143.3 °C (290 °F) with the combined logarithmic standard deviation of 0.137 to be reasonable and acceptable.

The fragility curve obtained from the linear approximation is judged to be more conservative in the low-pressure region, say, below 965.26 kPa (140 psia), than the lognormally distributed fragility curve based on the combined beta method. Both methods, however, yield similar results around 999.74 kPa (145 psia) (3-percent failure probability). In the pressure range from 999.74 kPa (145 psia) to 1103.16 kPa (160 psia) failure probabilities computed using the combined beta method are higher than those used for the PRA (3-percent to 25-percent failure probability). However, most challenges from the postulated severe accidents for the System 80+ containment are confined to the pressure zone below 999.74 kPa (145 psia) except for the high-pressure RV DCH event. The high-pressure RV DCH event produces a pressure range from 682.95 kPa (99 psia) to 1041.10 kPa (151 psia). In this range, fewer than 2 percent of the events are above 999.74 kPa (145 psia). The net effect of using the linearly fitted fragility curve for PRA is to produce higher DCH CCFPs than those using a combined beta method. Therefore, the linearly fitted fragility curve for PRA is acceptable.

19.2.6.4 Evaluation of Localized Leakage

The containment function can be compromised if excessive leakage occurs before the computed containment capacity pressure is reached. The objective of the following evaluation is to ensure that significant localized leaks would not occur before reaching the containment capacity pressures. Above the design- allowable pressure, leakage from the containment can occur from buckling at the transition area due to high temperature and at penetrations due to high temperatures and high pressures. The leakage potential from thermal buckling and from containment penetrations is evaluated below:

(1) Thermal Buckling

In the design-basis evaluation, the ASME Code Service Level C limit (emergency condition) does not require the consideration of temperature loading. This is based on the premise that the temperature loadings associated with LOCAs are short lived and would not affect the behavior of the steel shell. The temperature loadings associated with the severe-accident sequences may last a number of days. Thus, the effect of severe-accident temperature loading needs to be evaluated to ensure that the expected compressive stresses at the transition region (along the entire periphery of the shell) do not lead to buckling of the containment shell, thereby, causing a loss of containment function. The staff asked ABB-CE to address the margin against buckling due to the severe-accident temperature loading in CESSAR-DC Section 19.11.3.1.2.1.

ABB-CE responded to the staff's request and investigated local buckling in this area using the following nominal loads: (1) internal pressure of 466.77 kPa (53 psig), (2) severe-accident temperature of 232.2 °C (450 °F), (3) dead weights, and (4) live loads. The internal pressure of 466.77 kPa (53 psig) and temperature of 232.2 °C (450 °F) is selected on the basis of the observation that the severe-accident events of an SBO with a dry cavity, and a large-break LOCA with a dry cavity, produce a consistently high temperature in the first 24 hours; the corresponding peak pressure during the first several days of these events is below 466.77 kPa (53 psig). The stability safety factor against buckling for this severe-accident condition was estimated as 1.9.

The thermally induced compressive stress encountered during the severe- accident condition is strain controlled, or strain limited, rather than load controlled. Load-controlled buckling is characterized by such load applications as external pressure and dead weight, that continue beyond instability into the post-buckling region, resulting in gross deformation and loss of function. Strain-controlled buckling is characterized by loads that are strain limited, such as thermal loads, so that when buckling occurs, the strain is accommodated and the load is relieved. The process is self-limiting so that deformations are controlled without immediate loss or impairment of function. The essential difference between load-controlled buckling and strain-controlled buckling is recognized in the ASME Code Case N-47, Appendix T-1500, by setting different design factors of safety for each case. The design factor of safety for load-controlled buckling is 3.0, consistent with the ASME Code Section III, whereas, for strain-controlled buckling, the design factor of safety is set at 1.67.

For the System 80+ containment vessel with a factor of safety of 1.9, it is highly unlikely that thermal buckling could impair the function of the vessel. The strain is limited and the material is ductile so that the shell will not rupture due to buckling. If there are no penetrations in the region of buckling that might distort appreciably, then there should be no reason for loss of containment pressure due to buckling. Since the region of thermal buckling is at the base and away from penetrations, this is acceptable.

(2) <u>Containment Penetrations</u>

The System 80+ has a 6.71 m (22-ft)-diameter equipment hatch, two 3.048 m (10-ft)-diameter personnel locks, and containment piping penetration assemblies to provide for the passage of process, service, sampling, and instrumentation pipe lines into the containment; electrical penetrations for power, control and instrumentation; and a fuel transfer tube for ingress and egress of fuel assemblies. Details of the containment penetrations are presented in CESSAR-DC Section 3.8.2.1.3. The penetrations have stiffeners to limit distortion, such as buckling or ovalization under severe-accident conditions. This is acceptable because the stiffened hatch has the allowable external pressure of 1,138.3 kPa (165.1 psia) at 143.3 °C (290 °F) and 1,102 kPa (159.9 psia) at 232.2 °C (450 °F) which is higher than the ASME Code Service Level C pressure, the acceptance criterion, as defined in Item 3.b, Section 19.2.6.2 of this chapter.

(3) <u>Containment Penetration Seals</u>

Seals around penetrations are designed to seat under internal containment pressurization to ensure minimal containment leakage at higher pressures. In CESSAR-DC Section 19.11.3.1.4, ABB-CE describes the design of electrical penetration assemblies (EPAs) and mechanical penetrations for the assumptions and conditions for severe accident.

In NUREG/CR-5334, SNL, "Severe Accident Testing of Electrical Penetration Assemblies," November 1989, showed that no leakage was detected from any of the three EPAs during severe-accident conditions. They are (1) D.G. O'Brien EPA, 182.7 °C, 1073.52 kPa (361 °F, 141 psig) for 10 days, (2) Westinghouse EPA, 204.4 °C, 521.93 kPa (400 °F, 61 psig) for 10 days, and (3) Conax EPA, 371.1 °C, 935.61 kPa (700 °F, 121 psig) for 10 days.

For the EPAs, ABB-CE stated that a temperature of 182.7 °C (361 °F) and pressure of 1073.52 kPa (141 psig) for 10 days bound all "wet" reactor cavity sequences and are generally representative of the low-probability "dry" cavity severe accident sequences. Therefore, ABB-CE considered the failure of EPAs remote.

For the mechanical penetrations, ABB-CE stated that the onset of rapid failure of seals occurred above 329.4 °C (625°F) provided the seal was constructed from either a ethylene-propylene (EP), neoprene, or silicone. Seal failure was defined as the inability of the seal to maintain a high (approximately 1135.56 kPa (150 psig)) containment pressure. Gradual degradation of these seals was noted at the temperature in the 148.8 to 204.4 °C (300 to 400 °F) range. The typical EP seals will require more than 20 hours to fail when exposed for a sustained period of high temperature. Silicone-based seals have even longer high-temperature stability. The capability of either sealant material is sufficient to guarantee containment integrity for periods of more than 1 day for all "wet" cavity sequences. These sequences constitute more than 90 percent of the severe-accident transients. Analyses performed by the staff using the MELCOR computer model showed that the temperatures do not exceed 260 °C (500 °F) for dry cavity sequences in the first 24 hours.

In CESSAR-DC Section 19.11.3.1.4, ABB-CE states that the intent of the penetration seal design is to ensure that the selected seal and mounting will provide a minimum of 1-day containment integrity. This intent will be accomplished by a combination of selecting high-quality and high-capability seals, protectively mounting the seal so that it is not directly exposed to the containment environment, and providing double seals (inner and outer) whenever possible. This is judged by the staff to be achievable based on the current penetration design and is, therefore, acceptable.

ABB-CE states that as a consequence of its design philosophy for the System 80+, seal failures will not cause a failure of the containment before 24 hours and that for all "wet" cavity sequences the seal capacity is higher the containment capacity. Dry containment sequences that do not result in basemat melt-through or containment overpressure are assumed to fail because of temperature degradation of the seals. In order to estimate the consequences of a containment seal failure, ABB-CE estimated the seal leakage area to be 92.9 cm^2 (0.1 ft^2).

The staff considers the treatment of penetration seals acceptable for severe accident conditions. ABB-CE incorporates its leakage area as a function of the containment failure mode. The staff reviews ABB-CE's source term estimates in Section 19.2.3.3.6 of this report.

19.2.6.5 Conclusion

The staff concludes that the design of the steel containment under severe accident phenomenology will comply with the deterministic CPGs of SECY-90-016. The conclusion is based on (1) the evaluation of capacity using ASME Code Level C Service Limit and a 3-D FEM analysis, (2) the realistic to pessimistic failure probability assessments for various pressure ranges, and (3) the due consideration of the effects of any potential localized leakage from thermal buckling at the transition area, at the penetrations, and at penetration seals.

The median pressure capacity should ensure that the containment would serve as a reliable barrier against uncontrolled release of fission products as long as the internal pressure generated by severe accident events does not exceed 1185.89 kPa (172 psia) at 143.3 °C (290 °F) or 1103.16 kPa (160 psia) at 232.2 °C (450 °F). On this basis, the staff considers that localized leakage from thermal buckling at the transition area, from the penetrations, and from penetration seals is duly accounted for.

Comparison of the current PRA fragility values from the linear approximation with the fragility curve obtained from the combined beta method based on the lognormal distribution shows that both methods achieve reasonable results. The linearly fitted fragility curve for PRA is acceptable because the net effect of using the linearly fitted fragility curve is to produce higher DCH CCFPs compared to the combined beta method. This is conservative and acceptable.

19.3 Shutdown Risk Evaluation

19.3.1 Introduction

As part of the certification process for the System 80+ standard plant design, the NRC requested, in accordance with SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issue and Their Relationship to Current Regulatory Requirements," January 12, 1990, the design certification applicant (ABB-CE) to perform a systematic examination of shutdown risk, including evaluation of specific System 80+ design features that minimize shutdown risk, quantification of the reliability of DHR systems, identification of any vulnerabilities introduced by new design features and consideration of fires and floods with the plant in modes other than full power. This was also identified as DSER Open Item 5.4.3.5-1 and DSER Open Item 20.2-13. The following discussion documents the staff's evaluation and basis for resolving these DSER open items.

ABB-CE evaluated the System 80+ design for risks associated with plant conditions in Mode 3 (hot standby), Mode 4 (hot shutdown), Mode 5 (cold shutdown), and Mode 6 (refueling). ABB-CE concluded that the System 80+ is engineered with features that enhance shutdown safety by: (1) deliberate system engineering, equipment specification and plant arrangements for shutdown operation, (2) mode dependent control logic that assists and limits operations, (3) instrumentation, displays and alarms that clearly portray plant status in each mode and, (4) procedural guidance and technical specifications that address important shutdown evolutions. The staff reviewed this submittal availing itself of insights from NUREG-1449, a number of studies from the international community documented in NRC Information Notice (IN) 91-54, "Foreign Experience Regarding Boron Dilution," September 6, 1991, and a probabilistic risk assessment (PRA) of shutdown and low-power operating modes for a PWR to screen for important accident sequences.

The purpose of the staff review is to ensure that the System 80+ design has appropriately addressed the shutdown risk concerns based on experience with operating plants, including appropriate vendor guidance for COL applicants in areas of outage planning and control, fire protection, and instrumentation. Design improvements and/or design modifications ABB-CE identified were reviewed to ensure insights from shutdown operation experiences were addressed and that the design improvements reduce the likelihood of core damage and enhance public health and safety. Also, the staff evaluated vulnerabilities that may result from new design features; DHR capability; treatment of fires and floods with plant in modes other than full power; and related technical findings discussed in NUREG-1449.

Therefore, the staff's proposed applicable regulation for shutdown risk is as follows:

The application for design certification must include a systematic examination of shutdown risk including an assessment of:

(1) specific design features that minimize shutdown risk;
(2) the reliability of DHR systems;
(3) vulnerabilities introduced by new design features; and
(4) fires and floods with the plant in modes other than full power.

These items are discussed in the sections below.

19.3.2 Design Features That Minimize Shutdown Risk

ABB-CE described the System 80+ design features that minimize shutdown risk in Section 7 of CESSAR-DC Appendix 19.8A, "Shutdown Risk Evaluation Report."

19.3.2.1 Shutdown Cooling System

The SCS consists of two electrically and physically independent divisions, each with 100-percent capacity. This redundancy will provide the operator with a standby SCS system if any component of the operating system fails to perform its function. This is an improvement over existing PWRs which have shared components such as the heat exchanger. The SCS is a dedicated system and performs only the DHR function. This is an improvement over operating PWRs where the residual heat removal (RHR) systems also perform the low-pressure SI functions for emergency core cooling. These improvements allow SCS maintenance to be performed in Modes 1 through 4, thus increasing the SDC availability during shutdown operation conditions. To reduce the likelihood of a loss of inventory, SCS suction valves are interlocked with the reactor system pressure to ensure that low-pressure piping is not exposed to full system pressure. However, even if the interlocks fail or are bypassed, the low-pressure portions of SCS piping are designed to withstand the full reactor pressure without rupture (ISLOCA challenge). This is an improvement over existing plants where low-pressure systems are not capable of withstanding full system pressure.

The following additional improvements have been made to the System 80+ SCS design that will increase resistance to and reduce the loss of DHR:

- elimination of automatic closure interlocks the SCS suction valves
- improved protection against SCS pump excessive flow conditions
- improved RCS level instrumentation at midloop
- elimination of loop seals in suction lines

19.3.2.2 Containment Spray System

The CSS consists of two redundant and independent trains, each having 100-percent cooling capability. The CSS pump and the SCS pump in the same division are of identical design and are interchangeable. These pumps are connected by piping and valves such that one pump can perform the other's intended function. Thus, the CSS also serves as an alternate DHR system.

19.3.2.3 Component Cooling Water System

The CCWS design has two redundant divisions, each having 100-percent cooling capacity. This redundancy provides continuous cooling for safety-related components and the flexibility for the COL holders to perform component repairs and maintenance without loss of the CCWS function.

19.3.2.4 Electrical Distribution System

The EDS design has two redundant safety divisions. Each division can be powered from the following four separated sources:

- switchyard interface I
- switchyard interface II
- a Class 1E EDG
- a non-safety-grade CTG

This arrangement allows redundant power supplies to be maintained available even during periods of electrical system maintenance.

19.3.2.5 Containment

The System 80+ design incorporates a large operating deck inside the containment. The design of the containment operating deck includes open floor spaces assigned for storage and to accommodate various maintenance activities during outages. Some open floor spaces are used for pre-staging and laydown of equipment in support of maintenance activities. The spaces provided for these activities combined with pre-staged support tools eliminate the need to transfer components through the equipment hatch to work spaces outside the containment, thus reducing the number of times the equipment hatch must be opened. This facilitates maintenance and recovery of containment integrity
in shutdown conditions.

19.3.3 Decay Heat Removal Capability and Alternate Decay Heat Removal

19.3.3.1 Reduced Inventory Operation

Reactor Water Level During Midloop Operation

While the RCS level is lowered to within the hot leg (midloop) to allow necessary maintenance and testing activities, the risk of loosing SDC is increased from the possibility of vortexing at the SCS pump suction. The ability to accurately measure the water level and water temperature, and to monitor the SCS status to ensure adequate core cooling, is particularly important. Instrumentation used for shutdown operation is discussed in Section 19.3.5 of this chapter.

Guidance to support reduced inventory operations has been developed for the COL applicant. The guidance prohibits operations directly affecting the RCS pressure boundary. Midloop operations are only performed with the reactor vessel head on so as to ensure the availability of the heated junction thermocouple (HJTC) level indication system. Maintenance activities are not performed on the SCS or the operable containment spray pump. Planning should be such that the duration of reduced inventory operations is minimized. Also, the guidance identifies initiators that lead to a loss of SDC flow such as a loss of ac power or valve misalignments. The System 80+ reduced inventory operational guidance provides important insights to the COL applicant for outage control and planning.

Air entrained in the SCS piping may create problems that hinder the ability to provide continued SDC during reduced inventory conditions. To address this concern, ABB-CE designed the SCS piping to each respective pump suction in a continuously downward sloping path, thereby creating a self-venting path with no high point areas and no loop seals.

The staff finds these provisions acceptable and finds that ABB-CE has appropriately addressed the concerns in NUREG-1449 regarding midloop operations.

19.3.3.2 Loss of DHR Capability

In the event that the SCS is lost, the CSS provides an alternate means for DHR. If both the SCS and CSS systems are lost, the SDS and the SI pumps can be used to perform a feed-and-bleed operation to maintain core cooling.

In addition to coping with a loss of DHR, the System 80+ design will provide alternate makeup capability to replenish RCS coolant boiloff. At least two means of adding inventory to the RCS will be available whenever the RCS is in a reduced inventory condition as indicated in Table 2.4-3 of CESSAR-DC Appendix 19.8A. Operating guidance will be provided to specify the makeup water source, ways to provide injection of water into the RCS, and the recommended strategy to be used. During Modes 5 and 6, if all normal methods of decay heat and inventory replenishment are lost, alternate makeup capacity can be provided using the CVCS charging pump or the boric acid makeup (BAMU) pump. The BAMU pump takes suction from the BAST.

The BAST water can be transferred to the CVCS using BAMU pumps or by a gravity feed bypass line around these pumps that allows the contents of the BAST to be delivered directly to the CVCS charging pump suction. As a result of the System 80+ probabilistic risk assessment (PRA) insights during shutdown operations, ABB-CE recommends that plant procedures be developed to prevent both BAMU pump and CVCS pump from being out of service at the same time during reduced RCS inventory conditions. This will further enhance alternate makeup capability of the System 80+ design. The staff considers this a COL action item and will ensure that proper procedures be developed by the COL applicant to prevent the BAMU pump and the CVCS pump from being out of service at the same time during reduced inventory conditions. This COL action item is discussed in Section 19.3.9.2 of this chapter and is part of COL Action Item 19.3.9.2-1.

Additionally, the staff specifically looks for passive DHR means that could be accomplished without ac power (SBO) during shutdown operation. ABB-CE states that the SITs could be made available because they have gravity feed capability. Inventory addition from the two SITs would provide approximately 1.68 m (5.5 ft) of borated water rise in the RCS level and can make up for approximately 3 hours of core boiloff (assuming 4 days after shutdown). However, ABB-CE did not identify operating procedures to utilize this method as a COL action item because the System 80+ design includes the use of the combustion gas turbine which provides necessary ac power to maintain DHR capability in an SBO event. The staff agrees that the use of passive DHR means is not required. However, the staff encourages COL applicants to consider the available passive DHR method in their planning and control for outage and refueling operations. The staff will review this method if it is made part of an outage plan.

The staff concludes that ABB-CE has presented acceptable ways to sustain core makeup, and required equipment will be made available to maintain core cooling in the event of a loss of SDC.

19.3.3.3 ECCS Recirculation

As pointed out in NUREG-1449, DHR capability in postaccident conditions (ECCS recirculation) could be lost if debris from maintenance activities prevented the water from draining to the containment sump or blocked the sump pump suction lines.

Following an accident, water introduced into the containment will drain into the HVT. Any debris greater than 3.81 cm (1.5 in.) in diameter in the containment will be prevented from entering the HVT by a vertical trash rack located at the entrance to the HVT. This vertical trash rack will help to impede the deposition of debris buildup on the screen surface. Debris less than 3.81 cm (1.5 in.) in diameter will be permitted to enter the tank. The HVT will function as a trap for solids, allowing debris that enters the tank to accumulate and settle on the bottom of the tank. The IRWST spillways will be located at a high location to ensure that most of the debris in the water settles to the bottom of the tank before the water spills over into the IRWST. Debris that remains suspended will make its way to the IRWST and will be prevented from entering the SIS suction piping by debris screens. These screens will filter out particles greater than 0.22 cm (0.09 in.) in diameter. The screen design will allow visual inspections to detect any corrosion or structural degradation during refueling outages.

The design of the System 80+ IRWST, HVT, and their associated debris-blocking devices offers reasonable assurance that recirculation will not be impeded by debris during postaccident conditions. However, the COL applicant's outage plans should include provisions to control debris from maintenance activities, and to preclude such practices which would impede ECCS recirculation as temporary covers. The staff will review the COL applicant's outage planning and control program to ensure that ECCS recirculation under postaccident conditions can be maintained.

19.3.3.4 Effects of PWR Upper Internals

In NUREG/CR-5820, "Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors," May 1992, the staff and its contractor analyzed the assumed loss of RHR with the vessel upper internals in place to examine the possibility of core uncovery from a lack of coolant circulation flow. Such conditions could occur during the flooding of the refueling pool cavity while preparing for fuel shuffling operations. Under these conditions, the vessel upper internals may provide sufficient hydraulic resistance to natural circulation flow between the refueling pool and the reactor, and may prevent the refueling water from cooling the core if the RHR cooling is lost. The staff asked ABB-CE to address this issue.

ABB-CE discussed the effects of PWR vessel upper internals in Section 2.10 of the System 80+ shutdown risk report.

Using conservative assumptions, ABB-CE estimated that the time to reach saturation was 35 minutes. It is expected that plant operators will be able to provide alternate core cooling using the CSS within 35 minutes. Procedural guidance will be given to the COL applicant, specifically the emergency operations guidelines, to address a loss of DHR during Mode 6 with the upper internals in place. In the event that the CSS pumps cannot be used as backup to the SCS pump, the CVCS charging pumps are available to provide makeup to at least match core boiloff. The charging pump can be throttled to match decay heat and can provide adequate flow for almost 12 hours before the BAST is depleted (assuming 4 days after shutdown). The staff considers this a COL action item and will ensure that the COL applicant develops procedures which require at least one CSS train to be available. This COL action item is discussed in Section 19.3.9.2 of this chapter and is part of COL Action Item 19.3.9.2-1.

19.3.4 Reactivity and Inventory Controls

19.3.4.1 Rapid Boron Dilution

ABB-CE discussed rapid boron dilution issues in Section 2.6 of the System 80+ shutdown risk report (CESSAR-DC Appendix 19.8A), including considerations discussed by the staff in NUREG-1449. Possible flow paths of unborated water that could result in a slug of water being injected into the RCS were also identified in Table 2.6.1 of the System 80+ shutdown risk report. ABB-CE stated that the only possible source of an unborated water slug is the DVI lines. The water volume for these lines is determined to be a maximum of 3.40 m^3 (120 ft^3). This diluted water slug is assumed to be injected into the reactor vessel via the DVI lines at the maximum flow rate with four SIS pumps operating in order to minimize the potential mixing with RCS water. One RCP is then assumed to start flushing the water slug through the core at approximately 116 percent of design flow. Analysis results indicate that maximum positive reactivity addition of the event is less than 3 percent. This reactivity insertion is less than the available shutdown margin of 6.5 percent. Therefore, the core will remain subcritical.

Also, ABB-CE addressed the scenario discussed in IN 91-54. As described in IN 91-54, this particular event starts with the highly borated reactor being deborated as part of the normal startup procedure. The reactor is at hot conditions with the RCPs running and the shutdown banks removed. Unborated water enters the RCS through the CVCS charging pumps. It is then assumed that a LOOP occurs, resulting in a reactor trip and RCPs trip. The CVCS charging pumps restart because they are powered by emergency diesels. These pumps will continue to inject unborated water into the RCS from the volume control tank (VCT) during plant recovery. This unborated water will not be entirely mixed with the RCS water because of low natural circulation flow during startup and it is assumed to collect on the bottom of the vessel. When offsite power is recovered, it is assumed that the operators will restart the RCPs and resume the startup process. The unborated water then passes through the core as a slug and the reactivity insertion may be sufficient to cause a significant power excursion, possibly leading to fuel damage.

ABB-CE stated that the event described above will not occur in the System 80+ design because the CVCS is not safety-related and its charging pumps are not powered from onsite Class 1E emergency sources (i.e., EDGs). The CVCS charging pumps are powered from the CTG in the event of a LOOP. Therefore, if a LOOP occurs, the reactor will trip, as will the CVCS charging pumps. The CVCS charging pumps will not automatically be brought back on line. To resume injection, the operator must manually load the charging pumps on the PNS buses (powered from the AAC source) when site power is provided by the onsite emergency ac power. Therefore, injection of unborated water into the RCS from the VCT during plant recovery is unlikely.

Also, ABB-CE stated that NUREG/CR-5368, "Reactivity Accidents," and the System 80+ design were used to provide the logic and assumptions to assess boron dilution events mentioned in NUREG/CR-0105, "Estimates of Inter Dose Equipment of 22 Target Organ," July 1978, as follows:

- NUREG/CR-5368 states that boron dilution as a result of injection or leakage of diluted water from accumulator tanks (e.g., SITs for System 80+) into the vessel is an "incredible" event. This finding remains applicable to the System 80+ design. The applicable TS for the System 80+ design include more conservative assumptions than those assumed in the analysis.

- LOCA with diluted ECCS water is a Mode 1 issue and is discussed in CESSAR-DC Chapter 15.

- Rod ejection is a reactivity accident and is not related to boron dilution. A rod ejection accident is discussed in CESSAR-DC Chapter 15.

The staff concludes that ABB-CE has appropriately addressed the boron dilution concerns raised in NUREG-1449.

19.3.4.2 Potential for Draining The RCS

The staff stated in NUREG-1449 that primary coolant water could be lost during Modes 2 through 6 if the RCS is pressurized and its temporary pressure boundary fails. Failures of the temporary pressure boundary includes the use of nozzle dams in PWRs, ICI seals, and other drain paths.

In the System 80+ shutdown risk report (CESSAR-DC Appendix 19.8A), ABB-CE discussed SG nozzle dam integrity in Section 2.3, potential draining paths of the RCS in Section 2.12, and applicable CESSAR-DC Chapter 15 accidents and LOCA analyses for low-power operations in Sections 4.0 and 5.0, respectively.

19.3.4.2.1 Steam Generator Nozzle Dam Integrity

PWR nozzle dams are often used during refueling outages to allow inspection of the SG tubes. The System 80+ nozzle dam will be installed in the cold legs first and then in the hot legs. Likewise, the nozzle dam will be removed from the hot legs first and then from the cold legs. This installation and removal process will minimize the time that both SG hot legs will be simultaneously blocked by nozzle dams and will maximize the time the SG will be available for reflux cooling in case DHR is lost.

The nozzle dam will fail if the RCS pressure exceeds the nozzle dam design pressure without a pressure vent/release pathway, thus creating a direct RCS drain path (LOCA) to the containment through an open SG primary manway. The System 80+ nozzle dams are designed to withstand an RCS pressure of 275.8 kPa (40 psia) as compared to a typical pressure of 138 - 173.4 kPa (20-25 psia). The procedural guidance for reduced inventory operations will require the pressurizer manway to be opened before the nozzle dams are installed. The opening of the pressurizer manway provides a vent pathway and prevents possible RCS pressurization from exceeding the nozzle dams design pressure, thus ensuring that reactor water is not lost as a result of a loss of DHR. The pressurizer manway vent path is free of restrictions that could create unfavorable back pressure conditions.

19.3.4.2.2 In-Core Instrument Seal Table and Reactor Cavity Seal

The staff asked ABB-CE to address RCS leakage as a result of the ICI operations during refueling, and the ability to safely restore spent fuel cooling and maintain core cooling following failure of the reactor cavity seal during Mode 6 operations.

The loss of inventory from ICI seal table activities is minimized because the System 80+ reduced inventory operational guidelines prohibit the ICI seal table operations during midloop condition with the reactor vessel head installed. The System 80+ ICI seal table design, arrangement, and replacement is identical to the System 80 design at the Palo Verde Nuclear Power Station. The ICI seal table is located in the refueling pool area at several feet higher than the reactor vessel flange elevation. The withdrawal of the ICI assemblies will only be performed after the refueling pool has been flooded for refueling process. Therefore, the System 80+ design does not require temporary thimble tube seals in the ICI assemblies because of the replacement process mentioned. The staff reviewed the System 80+ ICI seal table arrangement and replacement process, and concluded that the loss of inventory from the ICI thimble tube seal failures is not a concern.

ABB-CE stated that if the reactor cavity seal failed during the refueling process, the refueling water would drain down to the reactor vessel flange level and would not result in vortex formation and air entrainment to SCS suction pumps. Hence, drainage from reactor cavity seal failure is self-limiting and SDC is not interrupted. Analysis results indicate that it will take approximately 4 hours for water in the refueling pool to drain down to the reactor vessel flange level.

Refueling water leaks through a failed reactor cavity seal collect in the reactor cavity region. The collected water is directed to the HVT and returns to the IRWST through the spillways connected with the HVT. The water is, therefore, available for return to the reactor vessel through the DVI lines via SCS and CSS injections. These system alignments are accomplished from the MCR by means of some local and manual operator actions. Additionally, alternate RCS makeup can be provided using available BAMU pumps that take suction from the BAST.

The System 80+ instruments and detection devices are available to operators to monitor the refueling water level with detection alarms set at 3.08 cm (2 in.) below the nominal level. The refueling level monitoring system provides water level indications down to the reactor vessel flange and alarms are located in the MCR.

Analysis shows that refueling water level drops to the top of the spent fuel being transferred in approximately 80 minutes. To preclude uncovering the spent fuel assembly, the fuel assembly must be lowered below the reactor vessel flange level. The fuel assembly can be lowered either into the reactor vessel or the end of the refueling cavity area containing the transfer system upender and core support barrel. These locations contain sufficient water to cover the fuel. The spent-fuel safe-storage process can be accomplished using the refueling machine, which takes approximately 3 minutes. EOGs will be provided to COL applicant to respond to a reactor cavity seal failure event. The COL applicant will develop plant-specific EOPs based on the System 80+ EOGs.

The staff considers this a COL action item and will ensure that proper procedures have been implemented to prevent, detect, and mitigate inadvertent loss of coolant through a failed reactor cavity seal. This COL action item is discussed in Section 19.3.9.2 of this chapter and is part of COL Action Item 19.3.9.2-1.

19.3.4.3 Applicable CESSAR-DC Chapter 15 Analyses in Shutdown Modes

ABB-CE discusses applicable CESSAR-DC Chapter 15 accidents postulated to occur in shutdown operations in Section 4.0 of CESSAR-DC Appendix 19.8A and states that consequences from these postulated accidents are bounded by accidents analyzed for power operating conditions indicated in CESSAR-DC Chapter 15.

The following seven events discussed in CESSAR-DC Chapter 15 have been postulated to occur during shutdown modes.

Increase in Feedwater Flow and Decrease in Feedwater Temperature

The evaluation indicated that transients during shutdown would not create any new consequences beyond those of the full-power events discussed in CESSAR-DC Section 15.1.2. The minimum departure from nucleate boiling ratio (DNBR) for this event was found to be greater than 3 as compared to the required minimum DNBR of 1.24. The RCS temperature and pressure would not exceed 100 percent of the design because of low decay heat level.

Increase in Main Steam Flow and Inadvertent Opening of an SG Relief or Safety Valve

The evaluation indicated that transients during shutdown would not create any new consequences beyond those discussed in CESSAR-DC Section 15.1.4. The minimum transient for DNBR was found to be greater than 2 as compared to the required minimum DNBR of 1.24.

Steam System Piping Failures Inside and Outside Containment

The evaluation indicated that transients during shutdown would not create any new consequences beyond those discussed in CESSAR-DC Section 15.1.5. The RCS pressure would remain less than 110 percent of the design pressure and would not violate the pressure-temperature (P-T) limits for brittle fracture. The SG pressure also would remain less than 110 percent of the design pressure, thus ensuring the integrity of the secondary system. The 2-hour inhalation dosage at the exclusion area boundary did not exceed the acceptance criterion. The minimum transient for DNBR was found to be greater than 2 as compared to the required minimum DNBR of 1.24.

Loss of Condenser Vacuum (LOCV)

The evaluation indicated that transients during shutdown would not create any new consequences beyond those identified in CESSAR-DC Section 15.2.3. The RCS and SG would not approach 110 percent of the design pressures. The LOCV would not challenge the P-T limits for brittle fracture. The minimum DNBR for an LOCV event postulated to occur at the highest decay heat flux just after shutdown, with no RCPs in operation, was found to be greater than 9. The DNBR will increase to a larger value when the event is associated with decay heat levels 4 days after shutdown.

Main Steam Isolation Valve Closure and Loss of Normal Feedwater Flow

The comparison between the MSIV closure and LOCV events, discussed in CESSAR-DC Section 15.2.4, is applicable for shutdown conditions. For the LOCV event discussed in Section 4.2.3 of the shutdown report, ABB-CE assumed a much faster reduction in steam flow rate than in the MSIV event discussed in CESSAR-DC Section 15.2.4. Therefore, the consequences of the MSIV event are bounded by the LOCV event postulated to occur during shutdown conditions. These assumptions resulted in a minimum DNBR of 9.

Feedwater System Pipe Breaks

A feedwater system pipe break postulated to occur during shutdown conditions following a heatup event would result in less severe consequences than the same event discussed in CESSAR-DC Section 15.2.8 because of much lower initial reactor power level. A heatup event can be mitigated by the pressurizer safety valve or the SCS relief valve, the LTOP enable/disable temperatures, the MSSVs, and the EFWS. The mismatched energy between primary and secondary pressure for a heatup event also is much less than that for the event discussed in CESSAR-DC Section 15.2.8. Therefore, there will be no violations of the P-T limits for brittle fracture of the RCS and no approach to 110 percent of the RCS and SG design pressures. Departure from nucleate boiling is not a concern for a heatup event because of low initial core power level.

Steam Generator Tube Rupture

The evaluation indicated that transients during shutdown Mode 3 would not create any new consequences beyond those discussed in CESSAR-DC Section 15.6.3. The SCS safety-relief valves would maintain the P-T limits for the RCS and these valves would not be prematurely actuated. The SG pressure would not approach 110 percent of the design pressure. Fuel integrity would be maintained, and radiological release doses would be even less if the SGTR event were postulated to occur in Mode 4 or 5 with the RCS loops filled.

19.3.4.4 LOCA in Lower Power Operations

ABB-CE stated that LOCAs could occur during shutdown. These LOCAs are normally associated with low RCS temperatures and pressures, and are considered small-break LOCAs. In Section 5.0 of the shutdown report, ABB-CE stated that consequences from the postulated LOCAs are bounded by LOCAs analyzed for normal operating conditions and that fuel acceptance criteria will be maintained as required by 10 CFR 50.46.

In Mode 4, ABB-CE stated that the most limiting size and location for a worst-case small-break LOCA is a break in a DVI discharge leg. A DVI line break of 0.036 m^2 (0.4 ft^2) at this location will minimize injection flow into the RCS.

ABB-CE performed a sensitivity study for a postulated DVI line break in Mode 4 to determine the available time for operator actions. The study indicated that the core will be uncovered in approximately 7 minutes from the time of the break due to boiloff, and the operator action to initiate SI within 10 minutes will prevent the cladding temperature from exceeding 1204.4 °C (2200 °F). ABB-CE stated that ECCS acceptance criteria will not be violated if SI of at least one SI pump is initiated 10 minutes into the transient. As a result, ABB-CE changed the System 80+ TS to require automatic SIS initiation be operable in Mode 4 conditions to reduce dependence on operator actions for core makeup and cooling. The staff performed an independent calculation of the DVI line break and confirmed ABB-CE's estimate.

The staff finds these provisions acceptable and concludes that automatic actuation of the SI injection in Mode 4 conditions ensures that the core will be covered and core cooling will be maintained.

At a meeting on February 8, 1994, and in subsequent telephone conference calls, ABB-CE presented additional information regarding the issue of RCS draindown during Mode 5 operation. Rapid RCS draindown from inadvertent system misalignment or valve opening is more likely to occur than a large pipe-break event during shutdown operations. The staff agrees with ABB-CE that rapid draindown events are the limiting loss-of-inventory events for Mode 5, and that the likelihood of a large pipe break during shutdown operations is considered remote.

Rapid draindown events predominantly result from numerous operator errors in performing the required actions as well as failure to follow maintenance procedures. Nevertheless, considerable time is available for operators to respond and to close the containment. ABB-CE estimated CDF of these events to be less than 1×10^{-7}/year. The SCS LTOP valve failure is considered the most limiting event, which indicates that the RCS water will drop to the bottom of the hot leg in 26 minutes, boiloff to the top of active fuel in another 19 minutes, and damage the fuel in an additional 20 minutes. The operator would have 45 minutes to diagnose the event and initiate injection before the core is uncovered. To preclude uncontrolled offsite releases, the containment is expected to be closed within an hour of event initiation. In addition to a manual SI capability, ABB-CE recommended changes in TS to require RCS level instruments with indications and alarms in the MCR during Mode 5 operations to help operators in diagnosing the transient. Containment temperature and radiation alarms also are available for operators to diagnose the transient.

The staff considers this issue to be technically resolved. It will remain a confirmatory item, however, pending formal documentation (amendment of CESSAR-DC) of the information presented at the meeting and subsequent information received during telephone conference calls. This issue was identified as part of FSER Confirmatory Item 1.1-1 in the advanced version of this report. Subsequently, ABB-CE incorporated this information into the CESSAR-DC in Amendment V. This is acceptable.

19.3.5 Instrumentation and Control During Shutdown Operation

The staff stated in NUREG-1449 that inadequate instrumentation and incomplete operating procedures, especially during periods of reduced inventory operations, have contributed to several loss-of-shutdown-cooling events at operating plants. Consequently, the staff recommended that PWRs of advanced designs include enhanced instrumentation capabilities to enable the operator to continuously monitor key plant parameters during reduced inventory operations. Also, the operator must be able to detect the onset of a loss of DHR early enough that mitigating actions can be taken to restore shutdown capability. As a minimum, instrumentation should be available to provide visible and audible indications of abnormal conditions in reactor vessel level, temperature, and SCS heat-removal performance.

ABB-CE addressed instrumentation and control systems in Section 2.8 of the System 80+ shutdown risk report (CESSAR-DC Appendix 19.8A).

<u>Level Instrumentation</u>

The System 80+ design uses four sets of level instrumentation for monitoring RCS inventory during draindown and reduced inventory operations. This instrumentation consists of two differential-pressure (dP)-based level sensor systems, and two different HJTC systems. One pair of wide-range dP level instruments is provided to measure the RCS coolant level from the top of the pressurizer to below the bottom of the hot-leg level. One pair of narrow-range dP level sensors is provided to measure the RCS coolant between the DVI nozzle elevation and the junction of the SCS suction line with the RCS hot legs. The wide-range and the narrow-range dP instruments have separate taps connected to each SCS suction line. These will ensure the independence and redundancy of the dP instruments and will operate with or without the reactor vessel head in place.

In addition to the dP-based level instruments, there are two sets of HJTC systems for reactor vessel level measurement when the reactor vessel head is in place and the plant is in Mode 5 reduced inventory operations. The first system uses two ICC probes that are located inside the reactor vessel. The range of these probes extends from the reactor vessel head to the fuel alignment plate. A second HJTC system provides narrow-range level indication for midloop operations when the reactor vessel level is in the hot-leg region. This system is specifically designed to provide accurate level indications using thermocouple probes concentrated in the hot-leg region. The measurement of RCS water level by these probes is limited to those periods when the reactor vessel head is installed.

The HJTC systems will compensate for the flow gradient across the core associated with the operating SCS. The HJTC sensors will have an accuracy and response time that are consistent with the maximum draindown rate of the RCS. The HJTCs are designed so that instrument signal and power are transmitted on individual electrical conductors. Failure of one HJTC sensor will not result in a loss of signal from the remaining sensors. The RCS level indications will be displayed and alarmed in the MCR during reduced inventory operations.

The staff concludes that the instrument range overlap will provide the operators a continuous indication of RCS inventory. The diversity of the level instruments (dP and HJTCs) will ensure against common-mode failures that could result in loss of RCS level response information. For midloop operations, the refueling HJTC probes will provide accurate level measurements to within 1 inch of the vessel level. This accuracy requirement is necessary because there is a very narrow margin between the minimum RCS level to prevent SCS pump cavitation and the level required for installing and removing the SG nozzle dams.

On the basis of this discussion of the System 80+ level instrumentation design, the staff finds appropriate RCS level indication is provided for reduced RCS inventory conditions.

Temperature Instrumentation

The System 80+ uses several different sets of temperature instruments to monitor the RCS coolant temperature during shutdown operations. The instruments available for measuring the RCS temperature consist of CETs, resistance temperature detector (RTD) sensors in the SCS suction and return lines, hot-leg RTDs, and refueling water level instrument temperature sensors (HJTC probe only).

The CETs measure the temperature of the coolant as it exits the top of the core. The CETs are bottom-mounted instruments; consequently, the CETs are available for measuring coolant temperatures even when the reactor vessel head is removed, except during fuel shuffling operations.

The RTDs in the SCS suction and return lines, and the hot-leg RTDs are effective only when the system is operating and the RCS coolant is flowing past the temperature sensors. These sensors will become ineffective following a loss of SDC flow.

ABB-CE states that refueling HJTC probes are used primarily to provide both level and temperature indications during midloop operations. The HJTCs will be disconnected before the reactor vessel head is removed; therefore, temperature indications provided by the HJTC probes will not be available. The temperature sensors will have associated alarms in the MCR for indicating the onset of loss of SDC and temperature rise in the RCS.

On the basis of these temperature instrumentation designs, the staff finds that appropriate indication of RCS temperature is provided to the operator during shutdown operations, and the design is, therefore, acceptable.

Shutdown Cooling System Performance

In addition to level and temperature instrumentation, the System 80+ design includes instrumentation for monitoring SDC pump suction pressure and motor current, which can be used to indicate the onset of pump cavitation.

The following discrete indications and alarms of SCS performance and RCS instrumentation will be located in the MCR:

- SCS heat exchanger inlet temperature
- SCS heat exchanger outlet temperature

- pump header pressure
- pump motor current
- pressurizer level
- RCS level
- RCS pressure
- core exit temperature
- refueling cavity level

On the basis of this discussion, the staff concludes that appropriate indication will be provided to the operator during shutdown conditions for monitoring RCS inventory, temperature, and SCS performance to ensure core cooling capability.

19.3.6 Flooding and Fire Protection

In NUREG-1449, the staff stated that the safety significance of flooding or spills during shutdown depends on the equipment affected by the spills and that such spills are most often caused by human error. Plant activities during shutdown and refueling operations may increase fire hazards in safety-related systems essential to the plant's capability to maintain core cooling. Further, Appendix R to 10 CFR Part 50, "Fire Protection for Nuclear Power Facilities Operating Prior To January 1, 1979," and current NRC fire-protection philosophy do not address the capability of the plant design to protect safe-shutdown equipment from fire and floods during shutdown operations.

The System 80+ design requires flood protection from both external and internal sources for all structures, systems, and components whose failure could prevent safe shutdown of the plant or could result in an uncontrolled release of radioactivity. ABB-CE discussed significant water sources that can cause internal flooding and spills and may disable safe-shutdown equipment in Section 2.13.3 of the System 80+ shutdown risk report (CESSAR-DC Appendix 19.8A). The CCWS and the emergency feedwater storage tanks (EFWSTs) were identified as major sources of possible floods or spills because of valve failure or a break in the system piping.

19.3.6.1 Flooding Protection

Essential systems may be at a higher risk for failure due to flooding and spills during shutdown because of the various and interrelated maintenance activities that may be in progress simultaneously. Past events have involved, for example, spills from the CCWS, SWS, condensers, and refueling pool seals. ABB-CE addressed the issue of the loss of DHR as a consequence of spills and internal flooding that may disable components of the SDC system (SCS) in Section 2.13.1 of the System 80+ shutdown risk report. Section 3.4 of this report also contains an evaluation of flood protection., for both external and internal sources.

The System 80+ design emphasizes the elimination or minimization of potential flood sources within safety-related areas and provides a boundary of separation between redundant DHR systems as the means of flood protection.

The SWS and the CCWS heat exchangers used to remove decay heat from the RCS are located outside of the nuclear annex. The condenser circulating water system is also located outside of the nuclear annex. The location of these major sources of water (which could be potentially unlimited sources) outside of the nuclear annex reduces in-plant sources of flooding to that contained in the closed systems in the plant. ABB-CE has identified the following potential sources of flooding inside the nuclear annex:

Flood Source	Volume
component cooling water system	700 m³ (24,700 ft³)
in-containment refueling water storage tank	2065 m³ (72,958 ft³)
emergency feedwater system	1325 m³ (46,785 ft³)
fire protection system	2270 m³ (80,203 ft³)
chemical and volume control system	4560 m³ (161,075 ft³)

In CESSAR-DC Section 3.4.4, ABB-CE states that seismic Category I structures are designed with flood protection measures in accordance with RG 1.102. Flood barriers are integrated into the design to provide additional flood protection while minimizing the impact on maintenance accessibility. Floods are controlled in the plant by the divisional structural wall which serves as a barrier between redundant divisions of safe-shutdown systems and components, so that a single flooding event will not affect redundant safety systems. The lowest elevation of the structure wall has no door or other passage, and the limited penetrations through the lower wall are sealed against water. The COL applicant will perform an evaluation to ensure that all penetrations in seismic Category I structures below the external flood level are properly sealed to protect safety-related equipment.

The staff asked ABB-CE to provide monitors for indicating status of the flood doors in its review of flood protection for the shutdown risk evaluation. In response to this concern, ABB-CE revised the CESSAR-DC to state that the flood doors have sensors with open and close status displays at a central fire alarm station. The open/close status of the doors will be continuously monitored and the monitoring station will be continuously manned. The monitoring station is located in the security central alarm station (CAS). The CAS is continuously manned and monitored by security personnel. ABB-CE indicated to the staff that if an alarm indicates that a door is mispositioned, a security officer is sent out to investigate. Security procedures are the responsibility of the COL applicant as per Chapter 13 of the CESSAR-DC. Also, communications are provided between the MCR and the security CAS. Flood alarm information is also retrievable in the MCR. The staff reviewed ABB-CE's submittal and finds that the flood door monitoring system is acceptable (see Section 3.4.1 of this report).

As stated above, the primary means of flood control in the nuclear annex and RB subsphere is provided by the divisional wall which serves as a barrier between redundant trains of safe-shutdown systems and components. Each half of the spherical containment subsphere is compartmentalized to separate safe-shutdown components to the extent practical, while maintaining accessibility to the compartment. The subsphere, which houses the front-line safety systems, is compartmentalized into four quadrants, with two quadrants on either side of the divisional structural wall. Although flood barriers separate the quadrants, the capability of removing equipment is maintained. EFW pumps are located in separate compartments within the quadrants, and each compartment is protected by flood barriers. Penetrations are sealed and there are no doors in the divisional wall that separates the nuclear annex and RB subsphere up to El. 70+0 ft, which is the maximum internal flood. Safety-related electrical components are located at higher elevations so that floods will not affect components. Flood barriers also provide separation between electrical equipment and fluid mechanical systems at the lowest elevations within the nuclear annex. Curbs provide similar separation at higher elevations. The COL applicant will ensure that all seismic Category I structures are protected against flood damage.

The nuclear annex floor drainage systems are separated by division, and Safety Class 3 valves prevent backflow of water to areas containing safety-related equipment. Each subsphere quadrant has redundant sump pumps (Safety Class 3) and associated instrumentation, which are powered from the EDGs in the event of LOOP.

On the basis of its review, the staff has determined that the System 80+ design minimizes the potential flood sources within safety-related areas and provides a boundary of separation between redundant DHR systems to prevent potential sources of flooding from affecting the DHR function. Therefore, the staff concludes that ABB-CE has provided adequate measures to protect the System 80+ plant from flooding.

19.3.6.2 Fire Protection

NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," reported that 10 CFR Part 50 (Appendix R) and the current NRC fire-protection philosophy do not address shutdown and refueling conditions and the impact a fire may have on the plant's ability to remove decay heat and maintain reactor coolant temperature. The insights obtained from NUREG-1449 were utilized as part of the review of System 80+ fire protection during shutdown.

In Section 2.7.3 of the shutdown risk evaluation report (CESSAR-DC Appendix 19.8A), ABB-CE stated that a defense-in-depth philosophy is employed in the design of the fire-protection system in order to reduce overall shutdown risk from fire. The elements of the defense-in-depth philosophy are to prevent a fire from occurring and to promptly detect, suppress, and mitigate the consequences of any fire that should occur. The fire-protection features will be independent of other features or systems that are routinely taken out of service during shutdown. These elements are discussed in reverse order below.

To mitigate the consequences of a fire during shutdown, the two redundant divisions of the DHR systems are separated from each other with 3-hour-rated fire barriers. All penetrations within these barriers are sealed with assemblies that are qualified to maintain the integrity of the 3-hour rating.

Additionally, the System 80+ design provides interdivisional separation. Within each division, the containment spray pump and the SDC pump can be interchanged with each other to provide DHR. The SDC pump is separated physically and electrically from the containment spray pump by 3-hour-rated fire walls and fire doors. Additionally, the valve that connects the two systems is located in a separate fire area so that a fire involving either pump will not prevent the operator from switching over to the other pump. Both pumps are electrically isolated from each other in that each pump is powered separately from different safety-related buses.

In the event of a major fire in the MCR, the remote shutdown panel is capable of controlling both divisions of safety-related DHR systems. The remote shutdown panel is physically and electrically independent of the MCR and is available during reduced inventory and refueling conditions. Instrument and control power for safety-related DHR equipment would be transferred from the MCR to the remote shutdown panel. The MCR utilizes fiber-optics that transmits two identical digitized control signals to operate equipment. Therefore, it is highly unlikely that a fire in the MCR will cause spurious operation of equipment (except in cases of shorting control switches), because two simultaneous, identical digitized control signals are needed at the de-multiplexer for a control action to be taken at the field device (equipment). The impact a fire may have on control panel switches causing hot shorts will be evaluated on a plant-specific basis.

Inside containment there are no divisional fire barriers between redundant safe-shutdown equipment, as described in CESSAR-DC Section 9.5.1.2. ABB-CE has designed the containment so that the redundant safe-shutdown components such as instruments and valves will be separated to the extent practicable as stipulated in SECY-90-016. ABB-CE stated that redundant safe-shutdown divisions are generally located in separate hemispheres, the cables entering the containment are widely spaced around the perimeter of the containment, and that the only in situ combustible material inside the containment is insulation on cables that are not associated with safe-shutdown functions, and that they will not affect safe-shutdown equipment in a case of fire in the containment. Inside the containment and the annulus, ABB-CE has utilized 3-hour-rated mineral-insulated cable or its equivalent for safe-shutdown functions and 1-hour-rated penetrations are installed in the containment. ABB-CE also states that while currently there are no 3-hour-rated penetrations, 3-hour-rated penetrations will be used should they become available. The staff finds that the use of 1-hour-rated penetrations, with the commitment to use 3-hour-rated penetrations if they become available, is acceptable because of the low combustible loading in the vicinity of the penetrations inside the containment and the annulus, and the cables for redundant safe-shutdown divisions are widely spaced around the perimeter.

In CESSAR-DC Section 9.5.1.2, ABB-CE states that the HVAC system is designed to remove smoke and mitigate smoke migration beyond the area of origin in the event of fire. The HVAC systems in the subsphere building and control complex are required by TS to be operable during shutdown. The dedicated fans for smoke purge are designed to exhaust at a minimum of 945 L/min/m² (3 ft³/min/ft²) of floor area. The normal ventilation is designed to provide an air flow of 315 L/min/m² (1 ft³/min/ft²) of floor area or more. ABB-CE states that the layout of the ductwork ensures ventilation of all corners of the area as much as practical. The design as described provides a lower pressure in the division experiencing the fire which will prevent or significantly reduce the amount of smoke migration to other divisions.

ABB-CE states that fire-protection features required to detect and suppress fires are independent from other features or systems routinely taken out of service during shutdown modes of operation. In Section 2.7.3.2 of CESSAR-DC Appendix 19.8A, ABB-CE states that the System 80+ fire protection is not degraded or reduced during plant shutdown because there is no reason to breach the fire boundaries, interrupt the detection system, or impair the fire hose (standpipe) system. All of these systems are provided specifically for fire protection and are not shared with, or dependent on, any other systems or features. The System 80+ fire-protection program stipulates that the fire-protection systems shall be operable for all modes of operation, including low-power and shutdown operations.

A fully trained and equipped onsite fire brigade is available for fire-fighting activities. The fire brigade is available during refueling and shutdown activities.

To prevent fires, the COL applicant will develop administrative controls for all modes of operation. ABB-CE lists areas which will have increased combustible loading during shutdown conditions in Table 2.7-1 of CESSAR-DC Appendix 19.8A. The administrative controls include, but are not limited to, the following:

- control of combustible and flammable liquids
- control of combustible material
- housekeeping
- control of open flame and hot work

A detailed review of the administrative controls will be performed during the plant-specific licensing process as identified in COL Action Item 9.5.1.5.1.

On the basis of its review, the staff has determined that the System 80+ protection philosophy and design minimizes the potential fire sources within safety-related areas, provides fire-detection and suppression features, and provides a boundary of separation between redundant DHR systems to prevent potential sources of fire from affecting the DHR function. Therefore, the staff concludes that ABB-CE has provided adequate measures to protect the System 80+ plant from fire.

19.3.7 Containment Integrity

GL 88-17, "Loss of Decay Heat Removal," was issued to PWR licensees and requested, among other things, implementation of procedures and administrative controls that reasonably ensure that the containment will be closed before the time that reactor vessel water level would drop below the top of the active fuel following a loss of SDC under reduced inventory conditions. Containment closure was defined as a containment condition in which at least one integral barrier to the release of radioactive material is provided. This definition of containment closure effectively reduces the likelihood of a release while providing the flexibility to have the containment building open under appropriate conditions.

While the System 80+ is in Mode 5, with the RCS in reduced inventory, and Mode 6 during core alteration, or reduced inventory, containment closure is ensured by compliance with the System 80+ TS. TS 3.10.5 requires the equipment hatch closed and held in place by a minimum of four bolts, one door in each air lock closed, and each penetration providing direct access from the containment atmosphere to the outside atmosphere is either (1) closed by an isolation valve, blind flange, manual valve, or equivalent, or (2) exhausting through operable RB containment purge exhaust system high-efficiency particulate air filters and charcoal absorbers, and is capable of being closed by an operable containment purge and exhaust isolation system.

In GL 88-17, the staff defined a closed containment as one whose equipment hatch door is closed and held in place by a sufficient number of bolts so that no gaps exist in the sealing surface. In CESSAR-DC Appendix 19.8A under Section 2.5.3.2.2.3, ABB-CE states that four bolts are sufficient to secure the equipment hatch so that no visible gap can be seen between the seals and sealing surface. Equipment hatches, installed at operating plants, of a design similar to the one to be used in the System 80+ have met the intent of GL 88-17 by using four bolts. It is the staff's opinion that the containment closure can be demonstrated by either a visual inspection or a local leak rate test between the double seals in accordance with Appendix J to 10 CFR Part 50. The hatch is designed to be pressure seated. Thus, any increase in pressure inside the containment will act to seal the hatch. In addition any radiation leakage will go into the nuclear annex.

ABB-CE defined containment integrity during Mode 5, with the RCS in reduced inventory, and Mode 6 during core alteration, or reduced inventory, in TS 3.10.5. The staff concludes that the barriers noted in TS 3.10.5 sufficiently separate the containment atmosphere from the outside environment. This is acceptable pending a formal revision to the TS and EOGs that excludes the use of waterloop seals as a method of maintaining containment integrity (barrier). This issue was part of FSER Confirmatory Item 1.1-1 in the advanced version of this report. ABB-CE incorporated this into the CESSAR-DC in Amendment W.

When in Modes 5 and 6, with the RCS not in a reduced inventory, the operator is required by Appendix B of the EOGs to initiate containment closure immediately upon loss of all RHR. In CESSAR-DC Appendix 19.8A under Section 2.5.3.2.2, "System 80+ Containment Features," ABB-CE states that the closure time of the equipment hatch is less than 1 hour with or without ac power and that the air lock doors could be closed in less than 10 minutes.

ABB-CE analyzed the time to core uncovery resulting from the loss of SDC events, and found that time to reach saturation was approximately 11 minutes, and the core would be uncovered in another 55 minutes. The staff considers the time required to close the containment hatch to be the most limiting when trying to reestablish containment closure. The staff considers this a COL action item and will ensure that the containment can be closed within an hour of the initiating event. This is COL Action Item 19.3.7-1.

ABB-CE has incorporated guidance in the System 80+ EOGs to help the COL holders to establish procedures, using such available instrumentation as containment temperature, pressure, and radiation monitors, to expeditiously close the containment hatch in the loss of SDC event in Mode 5 other than reduced inventory operations.

The staff finds ABB-CE's approach acceptable for the containment closure in Mode 5 other than reduced inventory operations; ABB-CE has appropriately addressed the concerns and insights identified in NUREG-1449.

19.3.8 Shutdown Risk Insights

The staff reviewed ABB-CE's shutdown risk PRA for the System 80+ design. The study addressed CDF from internally initiated events in Modes 3, 4, 5, and 6; and vulnerabilities while operating the plant in modes other than full power. The staff also considered human reliability insights, important human actions, insights from uncertainty, importance, and sensitivities analyses. Details of the PRA insights for System 80+ are in CESSAR-DC Chapter 19 and Section 19.1.5 of this chapter.

The fundamental conclusion of the staff evaluation of the PRA-based insights for System 80+ shutdown operation is that there are no significant vulnerabilities that would require design changes. The following are considered important shutdown risk insights for the System 80+ design:

- During plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures, training, and operator knowledge of plant configuration. This issue is discussed in Section 19.3.6 of this chapter.

- During plant shutdown, the integrity of fire protection and flood barriers between areas in the same division, such as quadrants, where systems comprising the alternate shutdown success paths are located, should be maintained. This issue is discussed in Section 19.3.6.2 of this chapter.

19.3.9 Technical Findings in NUREG-1449

In NUREG-1449, the staff discussed the following issues that are especially important for shutdown operations:

- outage planning and control
- technical specifications
- fire protection
- instrumentation

19.3.9.1 Outage Planning and Control

In Section 2.1 of the System 80+ shutdown risk report, ABB-CE discussed the use of the operational support information (OSI) program to ensure that design features are effectively utilized in the operation of the plant and also provides COL holders a formal means to transfer design-related bases for operations, regulatory operational commitments, and related information. The OSI program integrates information from various interrelated areas, including the maintenance plan, the reliability assurance program (RAP), and as-procured equipment characteristics, to ensure that the COL holder can efficiently operate the plant within the design bases. The OSI program also includes operational guidance to support reduced inventory operations. The operational guidance to support reduced inventory incorporates requirements to support midloop operations, SCS flows, restrictions related to vortexing characteristics, and capability to diagnose abnormal operating conditions. The OSI program is intended to ensure that the information needed to develop a sound outage plan and technical input needed for training will be available to the COL holders.

The staff considers outage planning and control a plant-specific issue and will review and ensure that the COL holders have appropriately addressed the outage and planning program to improve low-power and shutdown operations. The guidelines for planning and controlling outages should include the following:

- an outage philosophy which includes safety as a primary consideration in outage planning and implementation
- organizations responsible for scheduling and overseeing the outage; provisions for an independent safety review team that would be assigned to perform final review and grant approval
- control procedures which address both the initial outage plan and all safety-significant changes to schedule
- provisions to ensure that all activities receive adequate resources
- provisions to ensure defense in depth during shutdown and ensure that safety margins are not reduced; an alternate or backup system must be made available if a safety system is removed from service
- provisions to ensure that all personnel involved in outage activities are adequately trained; this should include operator simulator training to the extent practicable; other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing

This is COL Action Item 19.3.9-1.

The staff finds ABB-CE's approach to outage and planning for the System 80+ design acceptable. The OSI program provides sufficient guidance with regard to proper precautions, restrictions, design features, and requirements to guide COL applicants in the development of their shutdown risk program.

19.3.9.2 Operator Training and Procedures

ABB-CE has developed the OSI program to help COL holders develop plant-specific procedures and requirements. The COL holders are expected to prepare training programs and procedures for normal, abnormal, and emergency operations using guidance developed by the plant designer. The staff considers this a COL action item and will ensure that such important areas as DHR capability, inventory control (including LOCAs), electrical power availability, reactivity, and containment integrity have been properly addressed.

Additionally, the staff also considers it the COL applicant's responsibility to ensure the availability of the following procedures utilizing (1) the CVCS and BAMU pump capability to provide alternate coolant makeup during Modes 5 and 6 (detailed discussion of this issue is in Section 19.3.3.2 of this chapter), (2) the availability of the CVCS, SCS, and BAST to support Mode 6 operation with upper internals in place (detailed discussion of this issue is in Section 19.3.3.4 of this chapter), and (3) the availability of the SCS, CSS, and BAMU pump to respond to a failed reactor cavity seal event (detailed discussion of this issue is in Section 19.3.4.2.2 of this chapter).

This is COL Action Item 19.3.9-2.

19.3.9.3 Technical Specifications (TS)

ABB-CE established systematic requirements for reduced RCS inventory operations. The staff reviewed the System 80+ shutdown TS, using insights from technical findings discussed in NUREG-1449 and the staff's proposed model TS improvements to enhance the safe-shutdown operation of all nuclear plants which were discussed with industry during the open meeting with the owners groups on the improved standard technical specifications (STS) in July 1993 (memorandum from C. Grimes to B. Grimes, September 27, 1993). In Table 19.5, the staff tabulates additional limiting conditions proposed by ABB-CE, beyond those currently listed in the improved STS (NUREG-1432), for operation during reduced inventory. The System 80+ shutdown TS reflect redundant onsite ac power sources, one offsite ac power source, redundant SCS systems, independent means of monitoring of RCS level and temperature, independent means of monitoring SDC performance during reduced inventory conditions, and associated support systems to ensure the DHR capability can be maintained and to minimize the loss of DHR from a loss of ac power.

The System 80+ shutdown TS closely follow the staff's guidance on the proposed model TS improvements. Therefore, the staff concludes that the proposed System 80+ shutdown TS include requirements needed for managing risk during shutdown operations.

19.3.10 Conclusion

Based on the above, the staff finds the System 80+ design and the System 80+ Shutdown Evaluation Risk Report (CESSAR-DC Appendix 19.8A) acceptable, and meets the staff's proposed applicable regulation for shutdown risk. Further, the staff concludes that ABB-CE has adequately addressed the shutdown risk concerns identified in NUREG-1449 and has demonstrated that the System 80+ design will not introduce significant risk during shutdown operations.

Table 19.5 Comparison of limiting conditions in System 80+ TS and System 80 TS for operation during reduced RCS inventory

System 80+ Shutdown TS	Modes	System 80 General TS	Modes
Instrumentation * Two independent means of monitoring RCS level and temperature * Two independent means of monitoring SCS performance	5 and 6 with water level <120 ft	No specific requirements for reduced inventory operations	
Vent paths * An RCS vent path of ≥ [pressurizer manway removal] is established and maintained	5 with water level <117 ft) 6 with water level <117 ft) with reactor vessel head in place with one or more bolts tensioned	No specific requirements for reduced inventory operations	
Heat removal * Two SCS divisions operable * One containment spray pump operable	5 and 6 with water level <7.0 m (23 ft)	Shutdown cooling (SDC) - refueling operations • High water level condition: one SDC division operable • Low water level condition: two SDC divisions operable	6 with water level >7.0 m (23 ft) 6 with water level <7.0 m (23 ft)
Containment integrity * [The equipment hatch closed and held in place by a minimum of four bolts] * One door in each air lock closed * Containment penetration isolation using blind flange, isolation valves	5 and 6 with water level <117 ft	no specific requirements for reduced inventory operations	
Availability of ac power * One offsite power source * Two onsite power sources	5 and 6 with water level <117 ft	ac - shutdown * One offsite power source * One onsite power source	5 and 6

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

19.4.1 Introduction

In 10 CFR 50.34(f)(1)(i), the staff requires an applicant to "perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant." In accordance with 10 CFR 52.47(a)(1)(ii), ABB-CE has addressed 10 CFR 50.34(f)(1)(i) as documented in CESSAR-DC Appendix 19A. The staff's evaluation is presented below.

ABB-CE has made extensive use of the results of the PRA to arrive at a final System 80+ design. As a result, the estimated CDF and risk calculated for the System 80+ is very low, both relative to operating PWR plants and in absolute terms. The low CDF and risk for the System 80+ is a reflection of ABB-CE's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to CDF in previous PWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the System 80+ design. Among the improvements are a four-train dedicated safety EFWS which includes two turbine-driven pumps for increased availability in SBO sequences, a safety-grade reactor depressurization system for facilitating feed-and-bleed operation, a diverse and independent combustion gas turbine capable of providing ac power to any of the safety and non-safety divisions, an IRWST that eliminates the need for switchover from injection to recirculation following a LOCA, and an ac-independent system as a backup to the CSS. Several improvements have also been incorporated in the System 80+ design to mitigate the consequences of a core damage event. These include: an HMS capable of being powered from station batteries, a reactor CFS to enhance the potential for ex-vessel debris coolability, the use of limestone concrete in the reactor cavity to extend the time to basemat penetration, and a high containment ultimate pressure capacity to minimize the potential for early containment failure. These and other System 80+ design features which contribute to low CDF and low risk for the System 80+ are discussed in Section 19.1 of this chapter.

In response to 10 CFR 50.34(f)(1)(i), ABB-CE submitted an initial evaluation of potential System 80+ design improvements by letter dated April 24, 1992 (LD-92-056). On the basis of this evaluation, ABB-CE concluded that because of the small risk associated with the System 80+ design (estimated at approximately 330 person-rem over a 60-year plant life) none of the design improvements considered were cost beneficial.

This evaluation was subsequently revised in letters dated June 18, 1993 [LD-93-098], and September 30, 1993 [LD-93-143], to reflect (1) changes made to the Level 1 and 2 portions of the PRA, (2) consideration of an expanded set of design improvements, and (3) additional information regarding the basis for risk reduction and cost estimates for selected design improvements. The residual risk in the revised analyses (approximately 8 person-rem over a 60-year plant life) was substantially less than in the original analysis, and strengthened ABB-CE's original conclusion that none of the design improvements, beyond those already incorporated in the System 80+ design, were cost beneficial. Finally, ABB-CE incorporated their evaluation into CESSAR-DC Appendix 19A "Design Alternatives for the System 80+ Nuclear Power Plant."

In the advanced version of this report, the staff noted that ABB-CE will submit a final update to its evaluation of potential design changes in the next CESSAR-DC amendment to reflect changes in the modeling of SGTR sequences in the Level 1 and 2 PRA. The update will result in an increase in total risk to approximately 17 person-rem over a 60-year plant life, but will not change ABB-CE's original conclusion regarding the cost/benefit of any design alternatives. This issue was part of FSER Confirmatory Item 1.1-1. Subsequently, ABB-CE updated the CESSAR-DC in Amendment V with this information. This is acceptable and resolves this aspect of FSER Confirmatory Item 1.1-1.

19.4.2 Estimate of Risk for System 80+

19.4.2.1 ABB-CE Estimates

The results of the Level 2 portion of the PRA is a set of RCs, each with an associated source term and frequency of occurrence. In the Level 3 PRA, these source-term estimates are combined with meteorological data and population data to yield predictions on offsite radiological impacts. Total offsite consequences are obtained by weighing the consequences for each RC by the RC frequency.

In ABB-CE's PRA analysis, source terms for each RC were determined using a plant-specific version of the NRC-developed XSOR code. Offsite consequences were then calculated for each RC using the NRC-developed MACCS code. Consequences were determined for a reference site defined by the meteorological and population data reported in Revision 1 of the EPRI ALWR URD (Chapter 1, Appendix A, Annex B, Revision 1, May 1989). The ALWR reference site data were developed by EPRI to conservatively represent, that is, bound, the consequences at approximately 80 percent of the reactor sites in the United States.

ABB-CE's estimate of the cumulative offsite risk to the population within 50 miles of the site is approximately 17 person-rem, assuming a 60-year plant life. The small level of risk calculated by ABB-CE is primarily due to the low estimated CDF for the System 80+ (2×10^{-6}/reactor-year), combined with the relatively benign nature of the bulk of the releases from containment, in terms of timing and magnitude of release. As a result of the low estimated CDF and associated risk levels for the System 80+, any potential modifications which cost more than about $20,000 would not be cost effective, even if the design modification were to totally eliminate the severe accidents or their consequences.

19.4.2.2 Staff Review of ABB-CE Estimates

The staff has reviewed the major models and assumptions entered into ABB-CE's risk estimate. ABB-CE based its risk estimate on four major elements: (1) the mean value CDF estimate from the Level 1 PRA; (2) the MAAP computer code and supporting deterministic analyses for modeling accident progression, containment performance, and time and energy of release; (3) a plant-specific version of the XSOR code for estimating fission-product releases (source terms); and (4) the MACCS computer code, combined with meteorology and population data for a bounding reactor site, for estimating offsite consequences.

As discussed in Section 19.1 of this chapter, the staff finds the approach used by ABB-CE for assessing CDF and containment performance to be logical and sufficient for describing and quantifying potential core damage sequences. ABB-CE has also estimated the uncertainty inherent in the CDF estimate, which has been considered by the staff in assessing the merit of the design alternatives.

The staff notes that in addition to MAAP code calculations for containment performance, ABB-CE submitted additional analyses using the NRC-developed MELCOR code. The NRC staff has also performed a number of severe-accident confirmatory calculations, as described in Section 19.2 of this chapter. On the basis of ABB-CE and NRC calculations, the staff concludes that ABB-CE's characterization of accident progression and containment performance is acceptable.

The staff has reviewed ABB-CE's source term estimates for the major RCs and compared these predictions with estimates from NUREG-1150. The staff finds the two sets of estimates in reasonable agreement. Finally, the staff considers ABB-CE's use of the MACCS code in conjunction with the bounding site data in the EPRI requirements document to be an acceptable basis for estimating the consequences associated with severe-accident releases for the System 80+ design.

Considering the acceptability of ABB-CE's overall approach for quantifying the risk of severe accidents, the staff has based its assessment of the risk reduction potential for potential design improvements on ABB-CE's estimate of risk (17 person-rem over a 60-year plant life for internally initiated events). However, the validity of the conclusions of this analysis were tested by considering the uncertainties in CDF estimates, as well as the potential for core damage from external events.

19.4.3 Identification of Potential Design Improvements

19.4.3.1 List of Potential Design Improvements

ABB-CE identified a set of potential design improvements for the System 80+ based on (1) previous industry and NRC-sponsored studies of preventive and mitigative features which address severe accidents; (2) the dominant failure modes identified in the System 80+ PRA, including SGTRs; and (3) input from the System 80+ design engineering staff. Among previous studies considered are the SAMDA performed by the NRC staff for Limerick, Comanche Peak, and the advanced boiling-water reactor standard plant design, as well as the plant improvements explored as part of the NRC CPI program. Through this effort, ABB-CE developed a list of 63 potential design improvements. The list is presented in CESSAR-DC Table 19A.5-1 (Amendment V).

ABB-CE eliminated certain design improvements from further consideration on the basis that they are already incorporated into the System 80+ design. Examples of design improvements already included in the design are larger pressurizer and SGs, an SDS, an IRWST, an HMS, and a CTG. On the basis of this screening, 40 potential design improvements were retained for further consideration. The list of design improvements selected for further evaluation was presented in CESSAR-DC Table 19A.5-2 (Amendment V).

The staff has reviewed the set of potential design improvements identified by ABB-CE and finds it to be comprehensive. The list includes all improvements identified as part of the NRC CPI program, and the NRC review of SAMDAs for Limerick that would be applicable to the System 80+. The list also includes potential design improvements oriented toward reducing the risk from major contributors to risk for System 80+, specifically, SGTR events. The staff notes that the set of design improvements is not all inclusive, in that additional, perhaps less-expensive design improvements can be postulated. However, the benefits offered by any additional modifications would not likely exceed those for the modifications evaluated, and the costs of alternative improvements are not expected to be less than those of the least expensive improvements evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered. On this basis, the staff concludes that the set of potential design improvements identified by ABB-CE is acceptable.

The set of design improvements selected for further evaluation also appears to be reasonable. Among the improvements considered are a filtered containment vent and a flooded rubble bed core-retention device, two improvements specifically mentioned in NUREG-0660 for evaluation as part of TMI Item II.B.8. ABB-CE also considered a strategy to delay the time of reactor vessel failure by flooding the reactor cavity to a level above the reactor vessel lower head. Recognition of the need for this modification was instigated by the results of recent analyses of reactor vessel bottom head failure as documented in NUREG/CR-6056.

It should be noted that several of the improvements selected for further evaluation have been incorporated as part of the System 80+ design independent of this evaluation of design improvements. These include the following:

- RCP Seal Cooling — A dedicated, positive-displacement seal injection pump (air-cooled) independent of CCW is provided in the System 80+ design, thereby reducing the potential for RCP seal LOCAs during loss of CCW events.

- Alternative SFWS — The SFWS has been modified so that it can be used as a backup to the EFW system.

- Nitrogen-16 (N-16) Monitors — N-16 monitors will be installed to assist the operators in identifying SGTR events.

- Alternative Containment Spray — An independent CSS has been added as a backup to the front-line CSS, so that frequency of late steam overpressure failures is reduced.

- Hydrogen Purge Line — The existing hydrogen purge line in the System 80+ design can be used to vent the containment to avoid late containment overpressure failures.

- Alternative Concrete Composition — The use of a limestone-based concrete in the reactor cavity is specified to provide increased resistance to core concrete attack and basemat penetration.

These items are identified as potential design improvements in the section below, but have not been further evaluated since they have already been incorporated into the plant design.

19.4.3.2 Description of Design Improvements

The design improvements selected by ABB-CE for cost-benefit evaluation are described in Section 5 of CESSAR-DC Appendix 19A (Amendment V). These improvements are listed below, grouped according to the general objective of the improvement. The numbers in parentheses correspond to the design alternative number in the ABB-CE submittal.

<u>A: Increase Primary and Secondary Boundary Integrity</u>

- RCP Seal Cooling (A1) — Add a dedicated positive displacement pump for diverse seal injection that is not dependent on CCW, thereby reducing the potential for RCP seal LOCAs during loss-of-CCW events (added to design).

- 100-Percent SG Inspection (A2) — Perform eddy-current testing on 100 percent of the SG tubes each refueling outage in order to reduce the frequency of SGTR events.

- N-16 Monitors (A3) — Provide N-16 monitors to assist the operators in identifying SGTR events (added to design).

- Increase Secondary Side Pressure (A4) — Upgrade the design pressure of the secondary system, including the MSSVs, from 1,200 psia to 1,500 psia in order to reduce the frequency of SGTR events.

- Passive Secondary-Side Coolers (A5) — Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink to reduce the potential for core damage due to loss-of-feedwater events.

- Secondary Side Guard Pipes (A6) — Install guard pipes around the secondary piping between the containment and MSIVs in order to reduce the potential for multiple SGTRs given an MSLB.

- Improved Overpressure Protection (A7) — Incorporate modifications to the original System 80+ design, as described in CESSAR-DC Appendix 5E, to reduce the challenges and risk from ISLOCAs (added to design).

- Digital Large-Break LOCA Protection (A8) — Upgrade plant instrumentation and logic to improve the capability to identify symptoms/precursors of a large-break LOCA (leak before break), thereby reducing the frequency of large-break LOCAs and MSLBs inside containment.

B: Increase Decay Heat Removal Reliability

- Alternative Batteries and EFWS (B1) — Increase the capacity of the EFWS-related batteries so that loss of DHR due to battery depletion is eliminated.

- 12-Hour Batteries (B2) — Increase the battery size to accommodate a 12-hour rather than 8-hour duty cycle, thereby reducing the probability of failure to recover offsite power before core damage.

- Alternative Pressurizer Auxiliary Spray (B3) — Increase the redundancy and diversity of the pressurizer spray valves and charging pump, so that failures of the auxiliary spray to successfully depressurize the primary system are eliminated in SGTR sequences.

- Alternative High-Pressure Safety Injection (HPSI) (B4) - Provide an alternative or improved HPSI system, so that all core-damage sequences involving HPSI failures are eliminated.

- Alternative RCS Depressurization (B5) — Increase the reliability and diversity of the safety depressurization valves so that all sequences in which SDS fails are essentially eliminated

- Diesel-Driven SI Pumps (B6) — Replace two of the electric SIS pumps with diesel-driven pumps to reduce CCF of all four pumps and the risk from SBO.

- Alternative SFWS (B7) — Modify the startup feedwater pump so that it can be used as a backup to the EFW system, including during SBO events (added to design).

- Extended IRWST Source (B8) — Provide a separate borated water storage tank and pump for refilling the IRWST, thereby reducing the potential for RWST depletion in unisolated SGTR events

C: Improve Electrical Power Reliability

- Third Diesel Generator (C1) — Add a third, swing DG to lower the probability of SBO events and provide improved operational flexibility.

- Tornado-protection for Combustion Turbine (C2) — Provide tornado protection for the gas turbine generator and associated support systems to prevent loss of the system due to tornado and high-wind events.

- Fuel Cells (C3) — Use fuel cells in lieu of conventional lead-acid batteries, thereby extending the availability of dc power.

- Hookup for Portable Generators (C4) — Provide temporary connections so that portable generators could be used to power the turbine-driven EFW pump after the station batteries are depleted.

D: ATWS and External Events

- Alternative ATWS Pressure Relief Valves (D1) — Provide a system of relief valves that can prevent equipment damage from a primary coolant pressure spike in an ATWS sequence.

- ATWS Injection System (D2) — Modify the RCP seal cooling system to inject boron using existing sources of boron and existing piping and valves.

- Diverse PPS (D3) — Provide a third, diverse PPS to resolve I&C diversity concerns and reduce the frequency of ATWS events.

- Increased Seismic Capacity (D4) — Modify the plant design, including containment and SG support design, to meet an HCLPF of twice the SSE (i.e., 0.6 g).

E: Reduce Radioactive Releases

- Alternative Containment Spray (E1) — Provide an independent CSS as a backup to the front-line CSS, so that frequency of late steam overpressure failures is reduced (added to design).

- Filtered Containment Vent (E2) — Add a filtered containment vent similar to the multi-venturi scrubbing systems implemented in some plants in Europe to eliminate the potential for late containment overpressure failures

- Alternative Concrete Composition (E3) — Use an advanced concrete composition in the reactor cavity or increase the thickness of the basemat concrete so that basemat melt-through is prevented (added to design).

- Reactor Vessel Exterior Cooling (E4) — Provide the capability to submerge the reactor vessel lower head in water during severe accidents in order to enhance heat removal from the lower head and prevent melt-through of the lower head.

- Alternative Hydrogen Igniters (E5) — Provide dedicated batteries for the HMS in order to improve system reliability and further reduce the potential for containment failure from hydrogen combustion .

- Passive Autocatalytic Recombiners (E6) — Provide passive autocatalytic recombiners in addition to the existing HMS to provide improved hydrogen control, particularly in SBO sequences.

- MSSV and ADV Scrubbing (E7) — Route the discharge from the MSSVs and ADVs through a structure where a water spray would condense the steam and remove most of the fission products, thereby reducing the consequences associated with SGTR.

- Alternative Containment Monitoring System (E8) — Improve the containment isolation valve position indication so that risk from containment bypass sequences and ISLOCAs is eliminated.

- Cavity Cooling (E9) — Modify the reactor cavity configuration and the flowpaths between the IRWST and reactor cavity so that heat from the reactor vessel lower head or ex-vessel core debris could be transported passively to the IRWST, thereby reducing the potential for reactor vessel failure, EVSEs, and core-concrete interactions.

- Venting the MSSV in Containment (E10) — Route the MSSV steam releases back into containment in order to minimize releases to the environment in SGTR events.

- Hydrogen Purge Line (E11) — Provide the capability to vent the containment to avoid late containment overpressure failures (added to design).

- Water-Cooled Rubble Bed (E12) — Provide a bed of refractory pebbles that would impede the flow of molten corium to the concrete drywell structures and increase the available heat transfer area, thereby enhancing debris coolability.

- Refractory-Lined Crucible (E13) — Provide a ceramic-lined crucible and cooling system in the reactor cavity in order to eliminate the potential for basemat melt-through.

- Vacuum Building (E14) — Provide a separate building/structure that would be normally maintained at a vacuum and would be connected to the primary containment boundary following an accident, thereby depressurizing the primary containment and further reducing emissions from severe accidents.

- Ribbed Containment (E15) — Add ribbing to the containment shell to reduce the potential for buckling due to containment vacuum conditions (i.e., reverse pressure loading).

19.4.4 Risk Reduction Potential of Design Improvements

19.4.4.1 ABB-CE Evaluation

ABB-CE used the reduction in cumulative risk of accidents occurring during the life of the plant as the basis for estimating the benefit that could be derived from plant improvements. ABB-CE developed estimates of risk reduction by determining the approximate effect of each modification on the frequency of the various RCs in the PRA. ABB-CE's basis for estimating the risk reduction for each design improvement is discussed in CESSAR-DC Section 19A.5 (Amendment V). ABB-CE's risk reduction estimates for each potential design improvement are reported in Table 19.6.

The staff reviewed ABB-CE's bases for estimating the risk reduction associated with the various design improvements. The staff notes that considerable judgment was exercised in estimating the risk reduction potential, but that the rationale and assumptions on which the risk reduction estimates are based are, in general, reasonable.

19.4.4.2 Staff Evaluation

In view of the small residual risk for the System 80+, rather than performing an independent assessment of the risk reduction potential of each System 80+ design improvement, the staff used a screening-type approach for identifying the most promising design improvements.

The set of potential design improvements was initially screened using a bounding assumption that each improvement would eliminate <u>all</u> of the risk from internally initiated events for the System 80+ (17 person-rem for the 60-year plant life). This approach tends to over estimate the benefits because the System 80+ risk profile reflects contributions from several unique types of sequences, for example, SBO, containment bypass, and LOCAs. An individual design improvement would generally reduce or eliminate some of these contributors, but would have no effect on others. Moreover, there are numerous and diverse modes of containment failure which must be dealt with to ensure containment integrity in a severe accident. Thus, a carefully selected set of plant improvements would

Table 19.6 Cost-Benefit comparison for potential design improvements

Potential Design Improvement	Estimated Cost ($ Million)	Averted Risk (Person-Rem)	Cost-Benefit Ratio (1000 Dollars/Person-Rem-Averted)	
			ABB-CE	Staff [1]
RCP Seal Cooling (A1)	NA (added)	---	---	---
100 percent Steam Generator Inspection (A2)	1.5	9.4	160	88
N-16 Monitors (A3)	NA (added)	---	---	---
Increase Secondary Side Pressure (A4)	Not Estimated [4]	---	---	---
Passive Secondary Side Coolers (A5)	Not Estimated [3]	---	---	---
Secondary Side Guard Pipes (A6)	1.1	0.04	28,000	65
Improved Overpressure Protection for ISLOCA (A7)	NA (added)[6]	---	---	---
Digital Large Break LOCA Protection (A8)	Not Estimated [7]	---	---	---
Alternative DC Batteries and EFWS (B1)	2	0.1	18,000	120
12-Hour Batteries (B2)	0.3	0.04	7,500	18
Alternative Pressurizer Auxiliary Spray (B3)	5	8.0	630	290
Alternative High Pressure Safety Injection (B4)	2.2	5.0	440	130
Alternative RCS Depressurization (B5)	0.5	0.9	550	29
Diesel Driven Safety Injection Pumps (B6)	2	5.0	400	120
Alternative Startup Feedwater System (B7)	NA (added)	---	---	---
Extended RWST Source (B8)	1	5.3	190	59
Third Diesel Generator (C1)	25	0.03	8.3×10^{-5}	1,500
Tornado Protection for Gas Turbine (C2)	3	0.10	30,000	180
Fuel Cells (C3)	2	0.1	18,000	120
Hookup for Portable Generators (C4)	0.01	0.1	90	0.6[2]
Alternative ATWS Pressure Relief Valves (D1)	1	0.06	17,000	59
ATWS Injection System (D2)	0.3	0.06	5,000	18
Diverse Plant Protection System (D3)	3	0.06	50,000	180
Increased Seismic Capacity (D4)	Not Estimated [10]	---	---	---
Alternative Containment Spray (E1)	1.5	0.4	3,800	88
Filtered Containment Vent (E2)	10	0.03	3.3×10^{-5}	590
Alternative Concrete Composition (E3)	5	0.3	17,000	290

Alternative Hydrogen Igniters (E4)	1	1.8	560	59
Reactor Vessel Exterior Cooling (E4)	2.5	1.8	1,400	150
Passive Autocatalytic Recombiners (E6)	0.8	1.8	440	47
MSSV and ADV Scrubbing (E7)	9.5	9.2	1,000	560
Alternative Containment Monitoring System (E8)	1	0.1	10,000	59
Cavity Cooling (E9)	0.05	1.8	28	2.9[2]
Venting the MSSV in Containment (E10)	Not Estimated [5]	---	---	---
Hydrogen Purge Line (E11)	NA (added)	---	---	---
Water Cooled Rubble Bed (E12)	19	0.3	63,000	1,100
Refractory Lined Crucible (E13)	108	0.3	3.6×10^{-5}	6,400
Vacuum Building (E14)	Not Estimated [8]	---	---	---
Ribbed Containment (E15)	Not Estimated [9]	---	---	---

1 Staff estimate based on ABB-CE cost estimate and the assumption that all risk (17 person-rem) is eliminated.

2 Further assessed by the staff and found to have a cost-benefit ratio comparable to ABB-CE's estimate

3 Judged by ABB-CE to require major changes in plant structures and high costs

4 Judged by ABB-CE to pose serious design drawbacks with limited benefits

5 Judged by ABB-CE to require a major redesign effort and pose serious design drawbacks

6 Further design improvements to address ISLOCA judged by ABB-CE to be unnecessary given improvements already incorporated in design, as documented in Appendix 5E to CESSAR-DC

7 Judged by ABB-CE to offer a negligible improvement in plant safety given the existing design features of NUPLEX 80+

8 Eliminated by ABB-CE on basis of high costs and ineffectiveness for bypass sequences which dominate System 80+ risk

9 Eliminated by ABB-CE on basis of high costs and unquantifiable (small) benefit

10 Further design improvements to address seismic events judged by ABB-CE to be unnecessary given existing seismic capabilities

generally be needed — each one acting on particular components of risk — to significantly reduce total risk.

For those potential design improvements whose cost-benefit ratio was found to be within a factor of 10 of the $1,000/person-rem-averted criterion in the screening assessment, a more design-specific assessment was subsequently performed. This is discussed further in Section 19.4.6 of this chapter.

19.4.5 Cost Impacts of Candidate Design Improvements

ABB-CE determined the approximate costs for each design improvement. The costing methodology and assumptions are described in CESSAR-DC Section 19A.3.2. The cost basis for each plant improvement is given in CESSAR-DC Section 19A.5, item by item.

ABB-CE indicated that the cost estimates represent the incremental costs that would be incurred in a new plant, not costs that would apply on a backfit basis. ABB-CE also indicated that the costs were intentionally biased on the low side, but that all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. However, the cost analyses conservatively neglected any annual costs associated with operation of the design improvements, including testing, maintenance, and training.

For modifications that reduce the CDF, ABB-CE reduced the costs of the design improvements by an amount proportional to the reduction in the present worth of the risk of averted onsite costs. Among onsite costs considered were replacement power at \$0.013/kwh differential cost, direct accident costs including onsite cleanup at \$2 billion, and the economic loss of the facility at \$1.4 billion. The resulting costs for each of the design improvements are given in Table 19.6.

The staff reviewed the bases for ABB-CE's cost estimates and finds them reasonable. For certain improvements, the staff also compared ABB-CE's cost estimates with estimates developed elsewhere for similar improvements, even though the bases for some of these cost estimates were different. The staff considered the cost estimates developed as part of (1) the evaluation of design improvements for GESSAR II (NUREG-0979, Supplement 4) and (2) the review of SAMDAs for Limerick and Comanche Peak (NUREG-0974 and -0775, respectively).

The staff noted a number of inconsistencies in the cost estimates; for example, ABB-CE's cost estimates for certain improvements, such as 12-hour batteries (\$300K), and reactor cavity cooling system (\$50K) were lower than expected, whereas the costs for other improvements were much higher than expected, such as alternative concrete composition (\$5 million) and refractory-lined crucible (\$108 million). Nevertheless, the staff views ABB-CE's approximate cost estimates as adequate, given the uncertainties surrounding the underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side, with which these costs were compared.

19.4.6 Cost-Benefit Comparison

A cost-benefit comparison was performed to determine whether any of the potential severe-accident design features could be justified. ABB-CE's cost-benefit estimates for each potential improvement are reported in Table 19.6. Consistent with current NRC practice (NUREG-3568), ABB-CE used a screening criterion of \$1,000/person-rem-averted to identify whether any of the design improvements could be cost effective. As shown in Table 19.6, the potential cost per averted person-rem ranges from about \$25,000 to \$830 million for the various suggested modifications according to ABB-CE evaluation, far exceeding the \$1,000/person-rem-averted criterion. On this basis, ABB-CE concluded that no additional modifications to the System 80+ design are warranted.

As mentioned in Section 19.4.4.2, the staff used a screening-type approach for identifying the most promising design improvements, and performed a more detailed assessment for only those design improvements whose cost-benefit ratio was found to be within an order of magnitude of the \$1,000 dollar/ person-rem criterion in the screening assessment. The factor of 10 is considered to provide ample margin to cover uncertainties in risk and cost estimates, given that, in general, estimates for these factors were conservatively estimated.

The set of potential design improvements was initially screened using a bounding assumption that each improvement would eliminate <u>all</u> of the risk from internally initiated events for the System 80+ (17 person-rem for the 60-year plant life). This approach tends to overestimate the benefits because the System 80+ risk profile reflects contributions from several unique types of sequences, for example, SBO, containment bypass, and LOCAs. On the basis of the initial screening, all but two of the potential design improvements have a cost-benefit ratio at least a factor of 10 greater than the \$1,000/person-rem-averted criterion, in spite of the significant conservatism in assessing risk reduction potential in the staff's analysis. For these improvements, a more design-specific assessment was performed. The two exceptions are:

- Hookup for Portable Generators (C4) — Provide temporary connections so that portable generators could be used to power the turbine-driven EFW pump after the station batteries are depleted.

- Cavity Cooling (E9) — Modify the reactor cavity configuration and the flow paths between the IRWST and reactor cavity so that heat from the reactor vessel lower head or ex-vessel core debris could be transported passively to the IRWST, thereby reducing the potential for reactor vessel failure, EVSEs, and core-concrete interactions.

The staff notes that for these two modifications, the assumption that all residual risk for the System 80+ design is eliminated is overly conservative, since these improvements will have little impact on the SGTR sequences that dominate risk for System 80+. ABB-CE's risk reduction estimates, which take into account the actual plant risk profile, are judged to be more appropriate for these options. ABB-CE's risk-reduction estimates for the portable generator hookup option assume complete elimination of all sequences in which EFW is lost after battery depletion. The risk- reduction estimates for the cavity flooding option assume complete elimination of reactor vessel melt-through, basemat attack, and steam explosions. On the basis of these assumptions, the cost-benefit ratio for the two options are a factor of 25 or more higher than the \$1,000/person-rem-averted criterion. Furthermore, the staff notes that at \$10,000 and \$50,000, respectively, these are lowest cost modifications evaluated by ABB-CE, and appear somewhat low. The staff concludes that these design options would not be cost beneficial, and also would not substantially reduce overall risk for the System 80+ design since the improvements would not have an impact on the sequences that dominate risk for System 80+.

For several of the potential design improvements, ABB-CE did not perform a detailed assessment of costs or risk reduction because ABB-CE judged that these improvements would involve serious design drawbacks, major redesign efforts, or extremely high costs, or would be relatively ineffective given features already incorporated in the System 80+ design. The staff considers ABB-CE's bases for excluding these improvements from further evaluation to be reasonable.

The staff notes that even though the System 80+ design is essentially a paper design, relatively large costs are still to be anticipated for many of the design improvements because they would involve first-of-a-kind engineering, and would need to be integrated within the existing design. In addition, the introduction of a new system will trigger a series of related requirements such as incremental training, procedural changes, and possible licensing requirements. These are all legitimate costs that require consideration in a comprehensive cost estimate. The staff concludes that none of the modifications evaluated would be cost effective given the low residual risk for the System 80+ and the $1,000/person-rem-averted criterion.

The staff has considered the robustness of this conclusion relative to a number of critical assumptions in the analysis, as described below. These involve the effect of uncertainties in estimating CDF, the use of alternative cost-benefit criteria, and the inclusion of external events within the scope of the analysis.

On the basis of uncertainty analyses performed by ABB-CE for the Level 1 portion of the PRA (see Section 19.1.3.1.3 of this Chapter), the 95th- percentile CDF is approximately 5×10^{-6}/per reactor year. This is about a factor of 3 higher than the mean value on which the cost-benefit analysis is based, but still very low compared to operating plants and also in absolute terms. Even if the benefits of the various design improvements were requantified on the basis of this upper bound value, none of the improvements would become cost beneficial. This would remain the case even if the cost-benefit criteria was also increased by a factor of 10 to $10,000/person-rem-averted.

If external events are included, the estimate of System 80+ risk could be one or possibly two orders of magnitude higher than considered in this analysis. However, even if they CDF were two orders of magnitude higher, any design modifications or combinations which cost more than $1.7 million would not be cost effective even if they completely eliminated all risk. On the basis of the ABB-CE analysis, those modifications which were estimated to cost less than $2 million have a relatively low risk reduction potential, and would generally eliminate only about 10 percent of the residual risk from internal events. The lower cost improvements are also not expected to be effective in eliminating most of the added risk from seismic events. As a result, none of these improvements are expected to be cost effective when their actual effectiveness in reducing risk is taken into account.

The staff concludes that given the significant margins in the results of the cost-benefit analysis, the findings of the analysis would be unchanged even considering the factors discussed above.

19.4.7 Conclusions

As discussed in Section 19.1 of this report, ABB-CE has made extensive use of the results of the PRA to arrive at a final System 80+ design. As a result, the estimated CDF and risk calculated for the System 80+ is very low both relative to operating plants and in absolute terms. The low CDF and risk for the System 80+ is a reflection of ABB-CE's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to CDF in previous PWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the System 80+ design. These include the provision of a four-train dedicated safety EFWS which includes two turbine-driven pumps for increased availability in SBO sequences, a safety-grade reactor depressurization system for facilitating feed-and-bleed operation, a diverse and independent combustion gas turbine capable of providing ac power to any of the safety and non-safety divisions, an IRWST that eliminates the need for switchover from injection to recirculation following a LOCA, and an ac-independent system as a backup to the CSS. Several improvements have also been incorporated in the System 80+ design to mitigate the consequences of a core-damage event: an HMS capable of being powered from station batteries, a reactor CFS to enhance the potential for ex-vessel debris coolability, the use of limestone concrete in the reactor cavity to extend the time to basemat penetration, and a high containment ultimate pressure capacity to minimize the potential for early containment failure. These and additional System 80+ design features which contribute to low CDF and risk for the System 80+ are discussed in Section 19.1 of this Chapter.

Because the System 80+ design already contains numerous plant features oriented toward reducing CDF and risk, the benefits and risk reduction potential of additional plant improvements is significantly reduced. This is true for both internally and externally initiated events. For example, the System 80+ seismic design basis (0.3 g SSE) has been shown to result in significant ability to withstand earthquakes well beyond the design basis, as characterized by a HCLPF value of about 0.7 g. Moreover, with the features already incorporated in the System 80+ design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques. Specifically, when CDFs of 1 in 100,000 or 1,000,000 years are estimated in a PRA, it is the areas of the PRA where modeling is least complete, or supporting data is sparse or even non-existent, that could actually be the more important contributors to risk. Areas not modeled or incompletely modeled include human reliability, sabotage, rare initiating events, construction or design errors, and systems interactions. Although improvements in the modeling of these areas may introduce additional contributors to CDF and risk, the staff does not expect that additional contributions would change anything in absolute terms.

In 10 CFR 50.34(f)(1)(i), the staff requires an applicant to perform a plant/site-specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. The staff concludes that the System 80+ PRA, and ABB-CE's use of the insights of this study to improve the design of the System 80+ meet this requirement. The staff concurs with the ABB-CE conclusion that none of the potential design modifications evaluated are justified based on cost-benefit considerations. It is further concluded that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations, because the estimated CDFs would remain very low on an absolute scale.

19.4.8 References

1. ABB-CE final assessment of Design Alternatives -- CESSAR-DC Appendix 19A (Amendment W).

Appendix 19A: Open, Confirmatory, and COL Action Items Identified as Unresolved in the System 80+ DSER

19A.1 Resolution of DSER Open Items

In preparing the DSER for the System 80+ PRA, the staff identified 51 open items. ABB-CE replied to all DSER items. The staff reviewed ABB-CE's responses and in many cases made a RAI. The review by the staff of ABB-CE's responses to the DSER open items, including responses to subsequent questions, found that the applicant satisfactorily addressed these issues. Therefore, the staff considers that all open items raised in the DSER are resolved. Closure of these DSER open items is summarized below.

<u>Open Item 19.1.1.1-1</u>: At the time the DSER was prepared, ABB-CE was updating its IRWST design. In this DSER open item, the staff stated that the PRA will be revised by ABB-CE to reflect the design change and that the likelihood for an unisolable IRWST leak due to a pipe break will be reevaluated. ABB-CE used the final IRWST design in the PRA-based SMA. The PRA-based SMA evaluated the likelihood of an unisolable IRWST leak and was found acceptably low. This open item is resolved.

<u>Open Item 19.1.2.1.1.2-1</u>: The staff asked ABB-CE to evaluate the potential impact of failure to open of one or more PSVs to prevent RCS overpressurization following an ATWS event. This evaluation was needed to determine the success criterion regarding the number of PSVs that must open to achieve successful RCS pressure relief during an ATWS event. In response to this open item, the applicant performed a series of ATWS transient analyses to evaluate the System 80+ response to an ATWS event as a function of the moderator temperature coefficient (MTC) and the number of PSVs that must open to mitigate the pressure transient. The results of these analyses, in conjunction with other ABB-CE analyses to determine the Level C stress limit pressure (i.e., the pressure level above which RCS integrity, or the operability of the systems needed for safe shutdown, can be jeopardized), were used to determine the number of PSVs that must open assuming different MTC values. To account for uncertainties in the deterministic "best estimate" analyses, and for PRA modeling purposes only, it was conservatively assumed that the Level C stress limit pressure is 3200 psia. ABB-CE modified the ATWS event tree model to reflect failure of PSVs to open. The staff found ABB-CE's conservative modeling of this issue acceptable. This open item is resolved.

Open Item 19.1.2.1.1.2-2: The staff asked ABB-CE to justify the success criterion for the cooling of the IRWST following an ATWS event with consequential SGTR (the same success criterion for IRWST cooling as for other less severe accidents was utilized). The concern was whether one train of the CSS is sufficient to successfully cool the IRWST following an ATWS event with a consequential SGTR and the loss of other heat removal systems. The concern was raised because the success criterion used for IRWST cooling was based on normal feed-and-bleed loads, while in an ATWS induced SGTR event the reactor power is higher and the reactor coolant is continuously lost though the ruptured SG tube. In response to this open item, ABB-CE performed a mass and energy balance calculation for the IRWST. The energy deposited to the IRWST during the first 4 hours of the accident, when the reactor power exceeds the energy removal capability of one CSS train, was estimated. The mass and energy removal mechanisms considered in the calculation included the CSS heat exchanger and the leak through the rupture (assumed to have an average value of 30 lb/sec during this time period). This calculation indicated that although the IRWST water temperature would rise to saturation and a small portion of the IRWST water (about 1 percent) would be lost through evaporation, in addition to the water lost through the break, the IRWST would still have plenty of inventory to permit successful feed-and-bleed cooling of the primary system (the calculation indicates that only about 10 percent of the IRWST inventory would be lost during the first 4 hours). The staff review found that this calculation justifies the success criterion for cooling the IRWST during an ATWS-induced SGTR event. This open item is resolved.

Open Item 19.1.2.1.1.2-3: The staff asked ABB-CE to provide documentation and related thermal-hydraulic calculations to confirm success criterion of only one train of containment spray for cooling the IRWST following an ATWS event with a stuck-open PSV. The concern was whether one CSS pump (and its associated heat exchanger) is sufficient to remove the heat load on the IRWST and thus prevent containment overpressurization and failure of the injection pumps that take suction from the IRWST. In responding to this open item, ABB-CE used the results of the calculation performed for Open Item 19.1.2.1.1.2-2. Since the water loss from the IRWST in this case is less than in the SGTR case, sufficient IRWST water would be available at the suction side of the injection pumps and the injection capability would not be compromised. In addition, although the total amount of energy added to the IRWST in this case could be larger than that for a design basis LOCA, the challenge to containment due to overpressurization would be less in this case because the RCS energy is released via the IRWST instead of directly to the containment atmosphere as is the case of a LOCA. Transient containment pressure calculations for a LOFW transient, without any heat removal from the IRWST, indicate that it would take about 40 hours before containment integrity is compromised. This proves that containment pressurization challenge is not likely in this case. The staff agrees with the logic in ABB-CE's response. This open item is resolved.

Open Item 19.1.2.1.1.3-1: The staff asked ABB-CE to provide documentation/analyses on the time it takes to reach SCS entry conditions following failure of the SIS during a small LOCA. The concern was whether core uncovery would start before the SCS entry conditions are reached when ASC is used to depressurize the primary system during a small LOCA with SI unavailable. ABB-CE performed best estimate analyses to determine the upper bound of a LOCA break size for which the core would remain covered in the absence of SI (only passive SIT injection was assumed). These analyses indicated that the core uncovery can be prevented for small LOCAs with break sizes of 28 cm^2 (0.03 ft^2) or less. Once SCS entry conditions are reached, the SCS pump is aligned to inject borated water from the IRWST into the RCS to replenish the lost inventory. Since small LOCAs were defined in the PRA as an effective break area of 46 cm^2 (0.05 ft^2) or less, ABB-CE redefined (in the updated PRA) a small LOCA as a LOCA with break size equal to 28 cm^2 (0.03 ft^2) or less. This resolved the question posed in this open item. However, due to this change, the break sizes of medium LOCAs were expanded to include the 28-46 cm^2 (0.03-0.05 ft^2) range. Since a medium LOCA is assumed in the PRA not to require secondary side heat removal (i.e., break size is sufficient for DHR via the break only), ABB-CE was subsequently asked to prepare an analysis showing that a 28 cm^2 (0.03 ft^2) break is sufficient for DHR via the break alone. ABB-CE responded with a best-estimate transient analysis (OPS-93-0784, October 1993) showing that this is the case. This open item is resolved.

Open Item 19.1.2.1.1.3-2: For an SGTR event, followed by failure of SIS, ASC using both the intact SG and the ruptured SG is considered as a practicable means of cooling the RCS down to SCS entry conditions. The staff requested ABB-CE to document the results of any supporting thermal-hydraulic analyses to substantiate the feasibility and estimated reliability of this operation. In response to this open item, ABB-CE performed best-estimate transient analyses (results and assumptions documented in OPS-93-0641, August 26, 1993; OPS-93-0814, October 7, 1993; and OPS-93-0986, November 19, 1993). These analyses, which take into account expected operator actions following applicable operating procedures, indicate that ASC is feasible if initiated within approximately 15 minutes after the failure of the SIS. Its reliability was estimated by taking into account operator failure to diagnose in time the need for ASC as well as operator failure to perform the required actions for successful ASC. The staff found that the analyses (and their results) support ABB-CE's response. This open item is resolved.

Open Item 19.1.2.1.1.3-3: The concern of this open item was the potential depletion of the IRWST water inventory in the sequence involving failure of SI followed by successful ASC and SCS injection. In the original event tree, no consideration was given to the need for long-term DHR or IRWST cooling, or whether the ruptured SG has been identified and isolated to terminate the leakage. If an unisolable leak path exists (e.g., stuck open MSSV), the water inventory of the IRWST could be depleted early. In response to this question, the applicant modified the SGTR event tree by expanding the above sequence to three sequences to include the effects of an unisolable SG leak and a failure to refill the IRWST. The staff found ABB-CE's response to this open item acceptable. This open item is resolved.

Open Item 19.1.2.1.1.3-4: The potential for a RCP seal LOCA during a SBO scenario was not included in the System 80+ PRA. Since this is contrary to what is assumed in many operating reactors, the staff asked ABB-CE to provide justification for this. ABB-CE maintains that the RCP seals used in the System 80+ design are not susceptible to seal LOCAs as a result of loss of cooling. The staff pointed out that the data, analyses, and test results submitted by ABB-CE do not provide conclusive evidence to support this position. ABB-CE subsequently performed a sensitivity analysis to determine the potential impact on the System 80+ CDF if RCP seal LOCAs were assumed to be possible when cooling to the seals is lost. Models used in the NUREG-1150 study for Westinghouse plants were adapted for use in this sensitivity analysis. Various values of the conditional RCP seal failure probability, given loss of cooling, were considered. This analysis indicated that the estimated CDF from internal events for the System 80+ design is not sensitive to the RCP seal failure probability following a SBO or loss of cooling water event. The same conclusion was reached for postulated fire and flood induced RCP seal LOCAs. This result is due to the reduced likelihood of SBO events and the improved reliability of RCP seal cooling for System 80+ as compared to operating reactor designs. Reduced SBO likelihood is due to the following features: (1) two physically separate and electrically independent switchyards, (2) turbine-generator runback capability, (3) addition of the non-safety CTG which is independent and diverse from the EDGs, and (4) EDGs are provided with dedicated 125-V dc batteries. Improved reliability of RCP seal cooling is due to the redundant and diverse systems that perform this function: (1) two separate and independent CCWS/SSWS divisions, (2) two redundant and divisionally separated charging pumps, and (3) a diverse (air cooled) positive displacement RCP seal cooling pump. For this reason, RCP seal LOCAs were not modeled in the System 80+ PRA. The capability and reliability of these important features will be ensured by the incorporation in the certified design of appropriate certification requirements, such as ITAACs, RAP, and TS. The staff found ABB-CE's response to this open item acceptable. This open item is resolved.

Open Item 19.1.2.1.1.3-5: The ATWS events considered in the System 80+ PRA are those initiated by loss of main feedwater with failure of turbine trip. The MTC following an ATWS is considered adverse if it is larger than -3 pcm/°F. An ATWS event, with existence of an adverse MTC, is considered to lead directly to core damage. This accident sequence was found, by the PRA evaluated in the DSER, to be a dominant contributor to the total CDF from internal events. The concern was that the conditional probability of having an adverse MTC, given an ATWS, was taken to be 0.01 with an EF of 1. Since there are uncertainties involved in estimating the critical MTC and, hence, the probability of having an adverse MTC, the staff asked that these be reflected in an uncertainty analysis to be performed for this sequence. In response to this open item, ABB-CE changed the EF to 7, the value used in the NUREG-1150 PRAs. In addition, ABB-CE argued that, in the case of the System 80+ design, an EF of 7 for the "adverse MTC" event is a conservative assumption. The staff agrees with the applicant's response. This open item is resolved.

Open Item 19.1.2.1.1.5-1: CCFs not treated in the System 80+ PRA, which was evaluated in the DSER, include (1) CCF of check valves, (2) miscalibration of water level sensors or flow transmitters, and (3) CCFs of components due to operator or technician errors during test or maintenance. In response to this open item, ABB-CE modified the fault trees to incorporate CCFs of check valves. CCF probabilities were calculated using state-of-the-art methods. Review by the staff found that CCFs of check valves were appropriately included in the modeling and quantification of the fault trees. ABB-CE also responded that common cause "instrument miscalibration" failures were already included in the "developed" events used for the PPSs. These "developed" events were taken from detailed System 80 PPS models in CEN-327-A and were quantified using the failure rate data in that report. ABB-CE submitted a copy of CEN-327-A along with a brief explanation of what the inputs to the "developed" events are. The staff's review verified ABB-CE's explanation. Regarding CCFs due to testing or maintenance errors, ABB-CE responded that they were already included in the general CCF rates for the components modeled in the fault trees. The staff determined that, indeed, this was the case. On the basis of these considerations, this open item is resolved.

Open Item 19.1.2.1.1.5-2: The staff asked ABB-CE to resolve a number of human reliability analysis (HRA) items. These items included missing events in the SGTR event tree and use of the same failure probabilities for a given action under very different circumstances. In response to this open item, ABB-CE made substantial changes to the System 80+ HRA to reflect NRC concerns. Specifically, Section 19.4 of the SSAR on "accident sequence determination" was modified to include a description of the standard operator actions for each initiating event based on the EOP guidelines. The revised SSAR Section 19.4 formed the basis for the identification and quantification of the operator errors modeled for each initiating event. Operator actions needed to control pressure during an SGTR event were added. The staff found that these changes are adequate to address the concerns associated with this open item. This open item is resolved.

Open Item 19.1.2.1.1.7-1: The staff asked ABB-CE to perform additional sensitivity/uncertainty/importance analyses to determine the sensitivity of the risk estimate to parameters that have significant uncertainties. These analyses were needed to gain insights about the design, to strengthen or remove doubts about certain aspects of the design, and to provide input for such certification-related programs as ITAACs, RAP, and Technical Specifications. In response to this open item, ABB-CE performed all analyses sought by the staff. The results are documented in Section 19.15 of the SSAR. The staff review found that these analyses have provided sufficient information and insights about the design so they can be used to support the various certification programs. This open item is resolved.

Open Item 19.1.2.1.1.8-1: In determining the unavailability of MOVs for the RDS, simple demand probabilities were used. However, it is believed that these probabilities are time-dependent since assumptions about testing and maintenance intervals significantly affect the demand failure rates for MOVs. The staff asked that ABB-CE either use time-dependent reliability techniques to determine these demand probabilities or should provide additional information/justification for the assumed probability in the PRA. In response to this open item, ABB-CE re-calculated the reliability of the RDS MOVs by using a time-dependent technique and appropriate assumptions. The staff found ABB-CE's response adequate. This open item is resolved.

Open Item 19.1.2.1.1.8-2: The staff sought further documentation to demonstrate that the new design features incorporated into the System 80+ design do not introduce new and significant failure modes. In response to this open item, ABB-CE identified several new potential failure modes, such as spurious draining of the IRWST into the cavity and a spurious opening of a RDS valve. ABB-CE evaluated each of these new potential failure modes and concluded that none of them is risk significant. The staff review, based on judgment, found that the ABB-CE's investigation and evaluation of this issue is adequate. This open item is resolved.

Open Item 19.1.2.1.2.1-1: The reactor cavity flood system (CFS) design was modified during the course of PRA preparation, and the final system design was not reflected in the version of the PRA reviewed at the DSER stage. In addition, inconsistencies were noted regarding whether the cavity would be wet or dry if containment sprays are available. ABB-CE subsequently revised the PRA to reflect the final cavity flooding system design, and submitted additional information regarding the flooding of the cavity due to containment sprays. The applicant's response adequately addresses the concerns raised in the DSER. This open item is resolved.

Open Item 19.1.2.1.2.1-2: In the original PRA, details on the data and split fractions used in the containment safeguard event tree (CSET) were not given. The availability of systems in the CSET and their effects on the PDSs were, therefore, not clear. Of particular concern was the high probability of wet cavity cases, and its impact on core-concrete interactions. In response to staff concerns, ABB-CE provided conditional probabilities of each of the containment safeguard states, and additional information regarding the success criterion and success probability for the CFS based on the final design of the system. This information supports the probability of dry cavity and CCI in the System 80+ PRA, and resolves this open item.

Open Item 19.1.2.1.2.2-1: A number of concerns were identified in the DSER regarding the validity of several of the PDS deletion rules. In response to this open item, ABB-CE changed the deletion rules in question and corrected other inconsistencies discussed in the DSER. The changes were reflected in the updated PRA, and resolve the concerns raised in the staff's earlier review.

Open Item 19.1.2.1.2.3-1: Some of the parameters that are important to defining fission product release (e.g., release point) were included in the PDS definition, but were not included in the containment event tree or the RC definition. As a result, PDSs with significantly different in-vessel releases to containment could be grouped in the same RC. In response to this concern, ABB-CE revised the CETs and the related supporting logic models to include top events for fission-product scrubbing for the various release paths (e.g., in-vessel and vaporization releases). Although there are still parameters that are important to fission product releases that are not included in the CET top events (e.g., the RCS leakage rate, which affects fission-product deposition in the RCS), the staff concludes that these changes acceptably resolve the concerns raised in the earlier review. This open item is resolved.

Open Item 19.1.2.1.2.3-2: In the DSER, the staff noted that several of the CET end states in the original PRA were not necessary because they were physically impossible, and that certain additional end-states may be needed. In response to this item, ABB-CE corrected the CET to eliminate physically impossible end states, and added additional end states to cover outcomes not originally modeled. After reviewing the updated CETs, the staff concludes that the problems noted with the original CETs have been eliminated. This open item is resolved.

Open Item 19.1.2.1.2.4-1: The frequency of containment isolation failure in the System 80+ PRA was taken directly from WASH-1400. In response to staff concerns regarding the applicability of this value to the System 80+ design, ABB-CE performed a plant-specific assessment of the probability for containment isolation failure due to piping penetration failure. The contribution to containment isolation failure from other important penetrations, such as electrical penetrations, equipment hatch, and personnel airlocks, was not included in this assessment, but is controlled by other regulatory requirements, including technical specifications and periodic leak testing under Appendix J. On these bases, the staff concludes that ABB-CE's treatment of containment isolation in the PRA is acceptable. This open item is resolved.

Open Item 19.1.2.1.2.4-2: A number of questions were raised in the DSER related to probability values used in quantifying the DCH logic model. Specifically, ABB-CE was asked to provide further justification for the probability values used to assess: debris retention in the reactor cavity (CAVTGEOM), temperature-induced failure of the RCS (HSINTACT), and failure to actuate the SDS. In the updated PRA, the use of the cavity retention parameter (CAVTGEOM) to reduce the probability of DCH failure has been eliminated. Values used for the latter two parameters have also been modified and further justified in the updated PRA. The staff concludes that the treatment of these issues in the updated PRA adequately resolves this open item.

Open Item 19.1.2.1.2.4-3: In the original PRA, credit was taken for the HMS in sequences in which the reactor vessel fails at high pressure. Since the HMS igniters may not be effective in preventing a large burn coincident with reactor vessel breach at high pressure, ABB-CE was asked to provide a further evaluation of the effectiveness of the HMS in these cases. In response, ABB-CE modified the System 80+ PRA supporting logic model for "early containment failure" in the updated PRA to include an unconditional hydrogen burn on vessel failure for sequences in which the vessel fails at high pressure. This response resolves this open item.

Open Item 19.1.2.1.2.4-4: In the original PRA, only the quasi-static pressure load from a hydrogen burn was included in ABB-CE's analysis. The potential for either global or local detonations was not addressed. In response to this concern, ABB-CE included hydrogen detonation as a potential cause for early containment failure within the supporting logic models in the updated PRA. ABB-CE modeled the potential for deflagration to detonation transition (DDT), but considered the potential for directly initiated detonations to be negligible due to a lack of high energy ignition sources. The staff considers ABB-CE's focus on DDT to be appropriate given the use of a deliberate ignition system in the System 80+ design, and the lack of any identified high energy ignition sources inside containment. The probability values associated with DDT are also reasonable, and based in part on a semi-quantitative ranking scheme that considers plant-specific containment characteristics. On the basis of a more complete modeling of detonation potential in the updated PRA, and the overall acceptability of these models, the staff considers this issue to be resolved.

Open Item 19.1.2.1.2.4-5: In several parts of the original PRA analysis, the probability of high RCS pressure at the time of core damage was erroneously used to represent the probability of high RCS pressure at the time of vessel breach. This error has been corrected in the updated PRA.

Open Item 19.1.2.1.2.4-6: In order to develop an understanding of advantages and disadvantages of the reactor cavity design features, the staff asked ABB-CE to provide an assessment of (1) the ability of the System 80+ design to accommodate the loads associated with alpha and rocket failure modes and (2) the impact of these challenges on the System 80+ risk profile.

In response to this open item, ABB-CE provided an assessment of the ability of the design to accommodate these loads in Section 19.11 of the updated PRA, and included these failure mechanisms in the early containment failure logic model. The updated PRA results indicate that these failure modes do not contribute significantly to the total early CCFP. This open item is resolved.

Open Item 19.1.2.1.2.4-7: In the original PRA, the assigned values for the probability of failing to recover containment heat removal appeared overly optimistic (e.g., 0.0 for sequences involving loss of power, and 0.01 for sequences involving failure of components outside containment). Recognizing that there are many factors that may hamper the ability to recover, the staff asked ABB-CE to provide further justification for the values used for non-recovery probability of containment heat removal. In the updated PRA, ABB-CE performed a more detailed containment spray recovery analysis, and modified the non-recovery probabilities accordingly. The staff finds the revised values reasonable and, on this basis, the open item is resolved.

Open Item 19.1.2.1.2.4-8: A number of issues related to the late hydrogen burn model were identified in the DSER, in particular (1) double credit for the presence of an early ignition source in the model and (2) apparent redundancy in the conditions/parameters considered to be necessary for a late hydrogen burn. The staff also indicated that additional justification is needed for the 0.5 probability value assumed for sequences in which the ignition source is unavailable until late in the sequence, and sequences without sprays. In response, ABB-CE modified the supporting logic model for late hydrogen burn, and provided additional justification for the modeling assumptions. This included converting the logic models to Boolean expressions to confirm that dependency conditions are handled correctly. The staff notes that the late hydrogen burn models used in the updated PRA, while somewhat complicated and difficult to trace, appear to be free of the double counting and inappropriate dependencies identified in the DSER. On this basis, this open item is resolved.

Open Item 19.1.2.1.2.4-9: In the original PRA, ABB-CE did not consider the potential for containment overpressure due to non-condensible gas generation. The significance of this challenge is augmented by the potential for higher temperatures during events with CCI, which tend to reduce containment strength. In the updated PRA, ABB-CE explicitly considers this challenge. According to the updated PRA, there is no credible potential for overpressure if long-term heat removal by spray is available. However, in wet cavity cases where containment sprays are not available, containment failure due to combined steaming and non-condensible gas generation is assumed to occur. (Dry cavity cases will generally lead to basemat melt-through before containment overpressure failure.) The staff finds that the updated PRA provides a reasonable representation of the potential for CCI and the effect associated with non-condensible gas generation. On this basis, the staff considers this DSER open item to be resolved.

Open Item 19.1.2.1.2.4-10: In the original PRA, ABB-CE did not consider the potential for core debris dispersal to the upper compartment to lead to containment overtemperature failure. Such a failure could occur as a result of corium impingement on the containment shell, or continued decay heat generation and combustible gas generation in the absence of water. This failure mechanism is considered in the updated PRA. The probability of containment failure due to debris impingement is addressed in the supporting logic model for early containment failure; the probability of containment overtemperature failure as a result of CCI with containment sprays unavailable is addressed in the supporting logic model for late containment failure (see Open Item 19.1.2.1.2.4-11). The staff has reviewed these revised models and the associated CCFP values and finds them to provide a reasonable representation of the potential for direct containment heating events, and the range of containment loads that can result from such events. On the basis of the more complete modeling provided in the updated PRA, the staff considers this DSER open item to be resolved.

Open Item 19.1.2.1.2.4-11: The potential for localized containment failure due to degradation of penetration materials at elevated temperatures was not modeled in the original PRA, but has been factored into the updated PRA. The probability of a containment penetration failure (under dry cavity conditions) is assumed to be 1×10^{-3} in the updated model. A low probability of penetration failure will be assured by a commitment under D-RAP that penetrations will be designed and sealant materials will be selected to ensure that the seal and mounting will provide a minimum of 1 day's containment integrity. The staff concludes that ABB-CE's treatment acceptably resolves this open item.

Open Item 19.1.2.1.2.4-12: In the original PRA, ABB-CE assumed that once the reactor cavity was flooded the core debris would be coolable and CCI would terminate in the cavity. Since experimental studies indicate that CCI can continue despite the existence of an overlying water pool, the staff questioned the validity of this assumption. In response to this item, ABB-CE revised the CCI model in the updated PRA. For the base case analysis in the updated PRA, ABB-CE assumed a 50-percent probability of achieving debris coolability given a wet cavity. The updated PRA also considers the potential for basemat melt-through in wet cavity cases, and assigned a 1 percent probability to this failure mode on the basis of high heat transfer rates to the water. The impact of these assumptions on PRA results was separately determined via sensitivity analyses. The staff concludes that ABB-CE's modeling changes and supporting analyses adequately address the concerns raised in the DSER. This open item is resolved.

Open Item 19.1.2.1.2.4-13: In the original PRA a complete basemat melt-through to the underlying stone or soil was assumed to not result in an atmospheric release and was treated as a "no containment failure" case. Since the consequences from this case would not be the same as a containment failure case, the staff asked ABB-CE to provide a further evaluation of the fission product releases associated with basemat melt-through and, if significant, establish an additional RC to cover this release mode. This concern has been addressed in the updated PRA by separately representing three possible modes of containment failure due to continued CCI: (1) penetration of the basemat into the underlying soil, (2) penetration of the basemat concrete into the ECCS pump room (subsphere), and (3) containment failure due to cavity wall collapse. These different release types are weighted on a frequency basis in the updated PRA to determine the release characteristics for a basemat melt-through. The staff finds the revised model and quantification acceptable, in that it provides a more complete representation of containment failure modes due to CCI and a more reasonable treatment of the fission product releases given these failures. On the basis of the improved modeling provided in the updated PRA, the staff considers this open item to be resolved.

Open Item 19.1.2.1.2.5-1: In the original PRA, the availability of SGs late in an accident was assessed based on the plant damage parameter "steam generator availability". The staff noted that because steam generator availability was determined at the time of core damage, there was no guarantee that the SG would continue to be available for a long period of time after core damage. Hence, ABB-CE was asked to reevaluate the availability of SGs late in accident sequences, and requantify the potential for revaporization release based on the revised model. According to ABB-CE, a mission time for EFW is set to 24 hours for all transients in the updated PRA. Since the transients for which the availability of EFW (or SGs) late in the event is of concern tend to be sequences which result in the onset of core damage within the first 8 hours, the use of a 24 hour mission time for these events provides reasonable assurance the SGs will remain available late in the sequence. The staff concludes that ABB-CE's treatment of revaporization release in the updated PRA, and supporting justification, acceptably resolves the issues raised in the DSER. This open item is resolved.

Open Item 19.1.2.1.2.5-2: The potential for late release of iodine from the IRWST and the reactor cavity water was not addressed in the original PRA. Because in-vessel fission-product releases are discharged to the IRWST in some accident sequences, ABB-CE was asked to evaluate the significance of late release of iodine from the IRWST and pools in the containment, and revise the PRA to reflect the results of this evaluation. The revaporization release supporting logic model in the updated PRA now includes the consideration of revolatization of iodine. However, the basic event affecting late iodine release is assigned a zero probability, to reflect the fact that the System 80+ design will have provisions for controlling the pH of the RCS and IRWST such that late iodine release will not occur. On the basis of these provisions, the staff finds ABB-CE's treatment of this issue in the PRA acceptable. This open item is resolved.

Open Item 19.1.2.1.2.7-1: A number of concerns regarding the containment ultimate pressure capability distribution used in the original PRA were identified in the DSER. As a result, ABB-CE was asked to provide further analyses and justification to support the containment ultimate pressure capability distribution. The following issues were to be addressed as part of this evaluation: (1) the locations and sizes of containment failure, (2) failure of containment penetrations, and (3) the effect of containment temperature. ABB-CE has performed further analyses of containment structural response and capabilities under severe-accident conditions, including the impact of each of the above items on containment pressure capacity. On the basis of these analyses, ABB-CE has modified the containment ultimate pressure capability distribution, and incorporated the revised distribution in the updated PRA. The staff has reviewed ABB-CE's analyses and finds them acceptable (see Section 19.2 of this report). On this basis, this open item is resolved.

Open Item 19.1.2.1.2.8-1: Because the original PRA submittal did not include supporting uncertainty or sensitivity analyses, ABB-CE was asked to perform and submit the results of quantitative sensitivity and/or uncertainty analyses which investigate the influence of key severe accident and containment performance issues and parameters on risk results. In response, ABB-CE modified the containment event trees and added decomposition event trees to more fully reflect the range of potential outcomes for phenomena containing significant uncertainties (e.g., DCH and CCI). ABB-CE also performed and submitted separate analyses of the sensitivity of containment performance and offsite doses to key models/assumptions in the Level 2 and Level 3 analyses. The staff finds that ABB-CE's treatment of issue uncertainty within the event trees, complemented by the sensitivity analyses, provides reasonable assurance that the PRA reflects the significance of key actions, events, and phenomena. This open item is resolved.

Open Item 19.1.2.1.3.2-1: In the original PRA, a single MAAP calculation was performed for each RC to determine the source term for the RC. Because the sequence selected for analysis was based solely on frequency rather than on release characteristics, the staff questioned whether the source terms derived through this process were representative or bounding of all PDSs in the individual RCs. ABB-CE was, therefore, asked to perform a closer examination of all the PDSs in the individual RCs, and provide additional justification that the sequences selected to represent each RC reasonably bound all sequences in the respective RC. In the updated PRA, ABB-CE has grouped the PDSs in each RC according to initiator, and selected the dominant PDS in each group to represent that group. Mean release fractions were generated for each PDS group in the RC using the S80SOR code. The weighted average of these release fractions was used to represent releases for each RC. The staff finds this approach acceptable. This open item is resolved.

Open Item 19.1.2.1.3.3-1: Because of the lack of detailed design information for the containment system, and the limitation of the code used for consequence calculation (CRAC2), ABB-CE assigned the release locations for the various RCs as either the top of containment building or at grade in the original PRA. Although these locations were judged to be reasonable, the staff asked ABB-CE to provide a further assessment of fission-product release locations, using the results of the reevaluation of the containment pressure capability (see Open Item 19.1.2.1.2.8-1) and the MACCS code. A more detailed accounting of the release location is made in the updated PRA, based on consideration of containment failure mode and type of containment bypass. The assumed release locations for the various sequences are reasonable and consistent with the containment layout. Accordingly, this open item is resolved.

Open Item 19.1.2.1.3.4-1: As noted in the DSER, the credit taken for fission- product removal in the original PRA appeared to be optimistic. Accordingly, ABB-CE was asked to provide further justification for the decontamination factors assumed for the various mitigation systems, including the containment spray, the cavity flood system, and fission-product retention in the auxiliary building. As part of the updated PRA submittal, ABB-CE provided additional information related to decontamination factors for the various mitigation systems, and comparisons between selected source-term estimates and equivalent source terms in NUREG-1150. The staff has reviewed the updated fission product release fractions for the various RCs, and the rationale for differences in magnitude of release between RCs. On the basis of this review, the staff finds ABB-CE's source terms, and therefore credit for fission product removal, to be reasonable. On this basis, the open item related to decontamination factors is resolved.

Open Item 19.1.2.1.3.5-1: Because of the considerable uncertainty associated with source-term determination, ABB-CE was asked to assess the impact of uncertainty in key source term issues on risk estimates. In response to this and other open items related to source terms, ABB-CE used the S80SOR code for calculating source terms in the updated PRA, instead of relying solely on MAAP. (S80SOR is a System 80+ version of the NRC-developed ZISOR parametric computer code for source term determination.) Although S80SOR is capable of predicting a source term distribution for each RC, ABB-CE based its source- term estimates on the mean release fractions obtained from S80SOR. Because of the large uncertainty band for release fractions predicted by the code, the mean value is much greater than the median for most radionuclide categories. ABB-CE also performed and submitted separate analyses of the sensitivity of containment performance and offsite doses to key models/assumptions in the Level 2 and Level 3 analyses. The staff concludes that ABB-CE's treatment of source terms, complemented by the sensitivity analyses, provides reasonable assurance that the PRA reflects the significance of important release characteristics and phenomena. On this basis, the staff finds that the issues raised in the DSER related to source-term uncertainty have been resolved. This open item is resolved.

Open Item 19.1.2.1.4.2-1: The staff indicated in the DSER that ABB-CE should provide estimates of the individual risk of early fatality and societal risk of latent cancers for the System 80+ design, as well as calculations for comparison with the proposed NRC large release goal of 1.0×10^{-6}/year. In the updated PRA, ABB-CE presents the CCDF for dose (probability of exceeding a given dose) at 0.8 km (0.5 mile), which permits comparison with the EPRI ALWR public safety goal as well as proposed NRC large release goals. ABB-CE has also provided estimates of the offsite doses for a reference site as part of its analysis of design alternatives evaluation under 10 CFR 50.34(f)(l)(i) requirements (see Open Item 19.1.2.1.6-1). The information provided adequately resolves this open item.

Open Item 19.1.2.1.5-1: The staff indicated in the DSER that ABB-CE should treat more thoroughly the loads associated with potentially significant containment challenges. The following loads were to be addressed in this assessment: (1) DCH, (2) early hydrogen combustion, (3) EVSEs, and (4) core- concrete interactions. Each of these challenges is provided in the updated PRA. This has been done through either modification of the supporting logic models to explicitly represent the potential for containment failure due to the challenge (as in the case of steam explosions), or the development and incorporation of decomposition event trees to represent the associated containment loads (as in the case of DCH). The staff has reviewed the modeling changes and finds that they reasonably represent the phenomena. On this basis, the DSER open item is resolved.

Open Item 19.1.2.1.6-1: At the time the DSER was being prepared, the staff had not completed its review of the 10 CFR 50.34(f)(l)(i) evaluation. The staff has completed its review, and it is documented in Section 19.4 of this chapter. This resolves the DSER open item.

Open Item 19.1.2.2.3-1: At the time the DSER was being prepared, ABB-CE was revising the seismic risk analysis for the System 80+ design. The staff requested that a key element of the revised analysis should be the presentation of fragility curves for the plant as well as for dominant core damage sequences. In addition, the staff requested that an estimate of the overall plant HCLPF should be included. Subsequently, with NRC's agreement, ABB-CE performed a PRA-based SMA instead of a seismic PRA. The PRA-based SMA provided similar information, regarding the capability of the design to withstand earthquakes, as the fragility curves (requested in this open item). The plant HCLPF value was calculated by the SMA. The staff reviewed the PRA-based SMA and found it acceptable. This open item is resolved.

Open Item 19.1.2.2.3-2: The staff requested that in the base case analysis performed for seismic sequences, low-acceleration cutoffs should be used in the fragility functions. Subsequently, with NRC's agreement, ABB-CE performed a PRA-based SMA instead of a seismic PRA. This automatically addressed the concern expressed in this open item. This open item is resolved.

Open Item 19.1.2.2.3-3: The staff requested that recovery from relay chatter should be examined carefully to determine whether reversing the initial failure event actually reverses all of the effects of the initial failure. ABB-CE responded that this will not be a concern since solid-state switching devices and electromechanical relays, resistant to relay chatter, will be used in the NUPLEX 80+ protection and control systems. This open item is resolved.

Open Item 19.1.2.2.6-1: The staff asked ABB-CE to perform a PRA-based analysis of internal fires to support the development of the needed design insights. To achieve this objective, the staff requested that the analysis include a logic development which is sufficiently detailed to show adequate success paths given probabilistically significant fires. In response to this open item, ABB-CE performed a "scoping" quantitative risk analysis and used it, in conjunction with a qualitative fire analysis, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. The "scoping" fire-risk analysis was based on two key assumptions: (1) integrity of the divisional separation between redundant safety-related equipment (this divisional separation, which is extended in addition to the nuclear annex in the SSWS/CCWS building, prevents fires from propagating from one division to the other) and (2) the conservative assumption that a fire which initiates a transient, causes all safety-related equipment to fail in the division where the fire occurred. This "scoping" analysis considers fires in the nuclear annex and the SSWS/CCWS building. Fires in the MCR or in the containment were examined separately using both qualitative and quantitative arguments. The reason that a detailed PRA was not performed is the lack of detailed design information regarding cable routing as well as the locations of fire-detection and fire-suppression systems (a detailed PRA involves modeling the propagation of fire, smoke, and hot gases between all the various fire areas containing safety-related equipment). However, the scoping fire- risk analysis provided enough information to conclude that the System 80+ design should result in a plant with superior capabilities to prevent and mitigate fires compared to operating nuclear power plants. The staff reviewed ABB-CE's initial fire-risk analysis and made several RAIs. By means of these RAIs, the staff asked the applicant to provide the following: (1) an evaluation of the potential for migration of smoke, hot gases, or fire suppressants into other fire areas where they could affect safe-shutdown capabilities, including operator actions; (2) an evaluation of the potential that fires in some areas (such as the intake structure, the transformer yard, and the turbine building) could affect systems used to bring the plant to a safe-shutdown condition, in addition to causing a plant trip; (3) an assessment of control room fires, including the capability of using the remote shutdown panel in case the NCR becomes unavailable; (4) a list of items that should be verified in ITAACs to ensure the integrity of the divisional separation as assumed in the fire-risk analysis (e.g., wall and fire-barrier integrity, as well as the requirements that any penetration in the divisional wall must meet); (5) an evaluation of loss of seal cooling due to a fire and resultant RCP seal LOCA; and (6) an investigation of the risk from potential fire sources inside the containment. ABB-CE submitted all the information requested in these RAIs (LD-93-100, June 25, 1993; OPS-93-0629, August 23, 1993; OPS-93-0998, November 29, 1993; OPS-93-1080 and OPS-93-1083, December 15, 1993). Although this information did not change, the original fire-risk analysis significantly provided some useful new safety insights about the design, as well as a better understanding of assumptions made in assessing the risk from internal fires. The staff found that the System 80+ fire-risk analysis met its objectives, that is, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs. This open item is resolved.

Open Item 19.1.2.2.6-2: The staff asked ABB-CE to perform a PRA-based analysis of internal floods to support the development of the needed design insights. To achieve this objective the staff requested that the analysis include a logic development which is sufficiently detailed to show adequate success paths given probabilistically significant floods. In response to this open item, ABB-CE performed a "scoping" quantitative risk analysis and used it, in conjunction with a qualitative flood analysis, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support certification requirements, such as ITAACs.
The scoping flood-risk analysis was based on two key assumptions: (1) integrity of the divisional separation between redundant safety-related equipment (this divisional separation, which is extended in addition to the nuclear annex in the SSWS/CCWS building, prevents floods from propagating from one division to the other) and (2) the conservative assumption that a flood which initiates a transient, causes all safety-related equipment in the division in which the flood occurred. This scoping analysis considers floods in the nuclear annex and the SSWS/CCWS building. The potential for flood propagation from the turbine building to the nuclear annex, where safety-related equipment is located, was also examined using both qualitative and quantitative arguments. A detailed PRA was not performed because of the lack of detailed design information needed to identify the potential flood sources and flood levels, such as pipe routing, flood curbs, and flood barriers (a detailed PRA involves modeling the propagation of flooding between all the various flood areas containing safety-related equipment). However, the scoping flood-risk analysis produced enough information to conclude that the System 80+ design should result in a plant with superior capabilities to prevent and mitigate floods compared to operating nuclear power plants. The staff reviewed ABB-CE's initial flood-risk analysis and made several RAIs. The staff requested detailed information on potential sources of internal floods by area, including their capacity, as well as information on the equipment that can be affected by the flood, existing flood barriers, and potential passageways. ABB-CE submitted this information (LD-93-100, June 25, 1993) and the staff found it sufficiently detailed to support assumptions made in the flood-risk analysis. The staff asked ABB-CE to investigate the potential that some floods, if left unmitigated, could impact multiple systems in both divisions. ABB-CE submitted an analysis (LD-93-100, June 25, 1993) showing that flood water from any of the applicable flood sources will be contained below elevation 70+0 ft (the elevation below which there are no doors or passageways in the wall separating the two divisions in the nuclear annex where safety-related equipment is located). The staff asked ABB-CE to address potential loss (due to the flood) of non-safety systems credited in the PRA, such as the charging pumps and the instrument air, in addition to the one division of safety-related systems. The objective of this RAI was to verify that the assumption of failure of one division of safety equipment, made in the scoping flood-risk analysis, bounds the risk for all potential internal floods and flood areas. In its response to this RAI (LD-93-100, June 25, 1993), ABB-CE states that non-safety equipment, credited in the PRA, is also divisionally separated. Therefore, all equipment in one division that was credited in the PRA was assumed failed in the scoping analysis. The staff asked ABB-CE to investigate the potential that a flood in the turbine building, in particular from an unisolable source of water, propagates to the nuclear annex (where safety equipment and the control complex are located). In its response to this RAI (LD-93-100, June 25, 1993, and OPS-93-0629, August 23, 1993), ABB-CE states that there are no sources of "unlimited" external flooding in the RB and concludes that consequential flooding of safety-related equipment from turbine building sources is prevented by the following design features: (1) plant grade below openings to safety-related structures, (2) openings to safety-related structures above the maximum flood level for the turbine building, and (3) site grade so that water would flow away from structures where safety-related equipment is located. The staff asked ABB-CE to submit a list of items that should be verified in ITAACs to ensure the integrity of the divisional separation as assumed in the flood-risk analysis (e.g., wall and flood barrier integrity as well as the requirements that any penetration in the divisional wall must meet). ABB-CE submitted this list. Finally, the staff asked ABB-CE to evaluate the potential loss of RCP seal cooling due to a flood and the resultant RCP seal LOCA. ABB-CE evaluated this issue and did not find it to be risk significant. The staff found that the System 80+ flood-risk analysis met its objectives, that is, to search for design vulnerabilities and to identify important safety insights and assumptions about the design needed to support

certification requirements, such as ITAACs. This open item is resolved.

Open Item 19.1.2.2.6-3: The staff asked ABB-CE to perform an importance analysis to extract design insights from the seismic portion of the analysis. Subsequently, with NRC's agreement, ABB-CE chose to perform a PRA-based SMA instead of a seismic PRA. Since an importance analysis is not possible without a PRA, ABB-CE used the SMA results, in conjunction with appropriate sensitivity analyses, to extract insights related to the plant's capability to withstand earthquakes. The staff found ABB-CE's response acceptable. This open item is resolved.

Open Item 19.1.2.3-1: The staff asked ABB-CE to submit a risk analysis for shutdown and low-power operation. ABB-CE submitted this analysis. After several RAIs and ABB-CE responses to such RAIs (documented in Appendix 19.8A of the SSAR), the staff found that the System 80+ shutdown analysis is of adequate quality and completeness so that it could be effectively used to identify vulnerabilities and extract safety insights about the design during shutdown operation. This open item is resolved.

Open Item 19.1.2.4-1: The staff asked ABB-CE to use PRA results and insights to prepare list(s) of items to be included as certification requirements, such as ITAAC and RAP requirements. ABB-CE submitted these lists and the staff found them adequate. This open item is resolved.

19A.2 Resolution of DSER Confirmatory Items

Confirmatory items in the DSER were defined as areas in which the staff and ABB-CE agreed on a proposed resolution to an open item; however, additional documentation was required. ABB-CE responded to all confirmatory items as agreed with the staff and these items were closed. The closure of these confirmatory items is summarized below.

Confirmatory Item 19.1.2.1.1.2-1: The staff asked for documentation showing that with no secondary cooling and no safety injection tanks (SITs), a single SIS pump can prevent core damage during a medium LOCA. ABB-CE performed a transient analysis using the MAAP computer code and submitted documentation showing that this statement is accurate.

Confirmatory Item 19.1.2.1.1.2-2: The staff requested that the success criterion for ASC during a small LOCA (i.e., the criterion that "all four SITs must inject borated water into the RCS during depressurization") be modeled in the fault trees. ABB-CE added this in the appropriate fault tree.

Confirmatory Item 19.1.2.1.1.3-1: The staff noticed that the event "failure of a PSV to reseat after opening" was not modeled in the event trees developed for "loss of main feedwater and other transients" (these events could be equivalent to small LOCAs). In the revised PRA, ABB-CE modified the affected event trees in accordance with the results of applicable transient analyses.

Confirmatory Item 19.1.2.1.1.3-2: The staff asked ABB-CE to report separately the SBO cutsets from the rest of the LOOP cutsets (the concern was whether any important SBO sequences have been overlooked given that the staff was able to identify one missing cutset). A detailed breakdown of the LOOP cutsets into blackout and non-blackout cutsets was submitted in ABB-CE Letter LD-92-113.

Confirmatory Item 19.1.2.1.1.3-3: The staff noticed that an important SBO cutset was missing. This involved LOOP, followed by CCF of the diesel generators and CCF of the turbine-driven EFW pumps. ABB-CE included the missing material in the revised PRA.

Confirmatory Item 19.1.2.1.1.4-1: The staff was unable to solve 2 of 67 top- level functional fault trees using the Integrated Reliability & Risk Analysis System (IRRAS) computer code. ABB-CE investigated these two fault trees and supplied new versions to the staff along with a discussion of what was changed and what effect this had on the risk profile.

Confirmatory Item 19.1.2.1.1.5-1: The staff asked ABB-CE to confirm that the 88 modularized events used in the fault trees are independent of one another. ABB-CE explained the process used in developing these "modularized" events (LD-92-113, November 18, 1992) and showed that these are independent of each other.

Confirmatory Item 19.1.2.1.1.6-1: The staff requested further clarification in order to better understand why the staff's audit calculations of some sequence frequency (especially with loss of CCW and loss of HVAC) were found to be substantially different from those submitted by ABB-CE in the PRA. ABB-CE responded that, since several changes were made in the updated PRA, all sequences were requantified. The staff reviewed the new calculations and found no discrepancies.

Confirmatory Item 19.1.2.1.5-1: The staff noted in the DSER that the original PRA documentation contained several errors and inconsistencies that required correction or clarification (see Footnotes 4, 7, 8, 9, and 10; and Section 19.1.2.1.2.4 in the DSER). In response to the staff's review, the inconsistencies and errors raised in the DSER have been either corrected or eliminated by modeling changes in the updated PRA.

Confirmatory Item 19.1.2.1.5-2: In the DSER, the staff noted that the starting times for tracking fission-product release were not consistent throughout the PRA and needed to be clarified. The tracking of fission products has been clarified in the updated PRA, and varies by initiator. The staff reviewed this information and found the tracking of the fission-product release for the various sequences adequate.

Confirmatory Item 19.1.2.2.2-1: The staff asked ABB-CE to modify the SSWS fault tree to include the SSWS intake structure blockage by tornado-generated debris. ABB-CE made this modification in the updated PRA.

Confirmatory Item 19.1.2.2.2-2: The PRA submittal used in preparing the DSER discussed tornado-induced blockage of the service water intake. This event leads to core damage, since the diesels fail and power is not recovered for a long time. However, it was assumed in the PRA that following this event the non-safety-grade combustion turbine could be used to recover ac power. ABB-CE corrected this mistake in the revised PRA.

19A.3 Resolution of COL Action Items

COL Action Item 19.1.2.2.2-1: The COL applicant must confirm the invulnerability of the intake structure to tornado-generated debris. This COL action item identified in the DSER has been redesignated for this report as COL Action Item 19-1.

COL Action Item 19.1.2.2.3-1: A systematic effort should be made to identify elements of the plant that do not appear in the internal events model, but that may affect the performance of systems in a seismic event (passive structures, etc.). This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-2.

COL Action Item 19.1.2.2.6-1: It will be necessary to factor site-specific spectra into the seismic analysis performed at certification. In addition, it is necessary to verify details of layout and anchorage of critical components, by reviewing construction drawings and by performing walkdowns. This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-12.

COL Action Item 19.1.2.4-1: All external hazards (e.g., external floods) should be examined at the COL stage by factoring site-specific information into previously analyzed external events and by performing screening analyses of external events that were deferred at the certification stage. This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-12.

COL Action Item 19.1.2.4-2: It will be necessary to verify details of internal fire analysis and the layout of critical components and fire- suppression systems by reviewing construction drawings and by performing walkdowns. This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-5.

COL Action Item 19.1.2.4-3: It will be necessary to verify interaction of potential internal flood sources and details of layout of critical components by reviewing construction drawings and by performing walkdowns. This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-5.

COL Action Item 19.1.2.4-4: The effect of fire suppression systems on the behavior of other systems will need to be examined at the COL stage. This COL action item identified in the DSER has been redesignated for this report as part of COL Action Item 19-5.

ABB-CE's responses are in agreement with the content of all of these COL items.

20 GENERIC ISSUES

Introduction

In this chapter, the staff discusses its evaluation of (1) the compliance of the ABB-Combustion Engineering (ABB-CE) System 80+ design with 10 CFR 52.47 (a)(1)(iv) and 52.47(a)(1)(ii), and (2) the incorporation of operating experience into the System 80+ design. The applicant for a standard design certification is required by 10 CFR 52.47(a)(1)(iv) to propose resolutions of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) defined in NUREG-0933, "A Prioritization of Generic Safety Issues," that are (1) technically relevant to the design and (2) identified in the applicable supplement to NUREG-0933. In addition, the applicant is required under 10 CFR 52.47(a)(1)(ii) to propose resolutions to the technically relevant portions of Three Mile Island (TMI) Action Plan items addressed in 10 CFR 50.34(f).

Because a large number of issues are relevant to the System 80+ design, the staff has grouped its evaluations into the following sections, according to the issue type in Appendix B of NUREG-0933:

- Section 20.2 contains the task action plan items.
- Section 20.3 contains the new generic issues.
- Section 20.4 contains the TMI Action Plan items.
- Section 20.5 contains the human factors issues.
- Section 20.6 lists the 50.34(f) TMI Action Plan items relevant to the System 80+ design.
- Section 20.7 discusses the incorporation of operating experience into the System 80+ design through generic communications.

20.1 Overview of Staff Conclusion

Compliance With 10 CFR 52.47(a)(1)(iv)

As stated above, an application for design certification must include proposed resolutions of those USIs and medium- and high-priority GSIs identified in the NUREG-0933 supplement that was current six months prior to the application, and which are technically relevant to the design.

By letters dated March 30 and August 21, 1989, ABB-CE applied for design certification of the System 80+ standardized nuclear power plant design in accordance with the provisions of 10 CFR 52.45. In the initial application dated March 30, 1989, ABB-CE applied for a design certification in accordance with Appendix O of 10 CFR Part 50. On August 21, 1989, ABB-CE revised its design certification application to be pursuant to 10 CFR Part 52. However, in its letter dated May 1, 1991, the Nuclear Regulatory Commission (NRC) staff stated that the ABB-CE application for the System 80+ design conformed to 10 CFR 52.47 and included Combustion Engineering Standard Safety Analysis Report—Design Certification (CESSAR-DC), as amended through Amendment I by ABB-CE's submittals dated April 26, July 12, and October 29, 1990, and March 4, 1991 (listed in Appendix A of this report). Therefore, to conform with 10 CFR 52.47 the applicable NUREG-0933 supplement is six months prior to ABB-CE's submittal of March 4, 1991. However, in Amendment U to CESSAR-DC Chapter 20, ABB-CE committed to address the relevant issues in Supplement 15 of NUREG-0933, dated December 31, 1992. The applicable supplement of NUREG-0933 is, therefore, Supplement 15.

The staff reviewed Supplement 15 to NUREG-0933 to identify the list of issues contained in Appendix B of NUREG-0933, "Applicability of NUREG-0933 Issues to Operating and Future Plants," that should be addressed to conform to Section 52.47(a)(1)(iv). In addition, the staff added five other issues (A-17, A-29, B-5, 29, and 82) that were resolved without the issuance of new requirements, but for which the staff had recommended the development of specific guidance for future plants.

The issues needed to comply with Section 52.47(a)(1)(iv) are evaluated in Sections 20.2 to 20.5 of this chapter. Additional issues that ABB-CE considered applicable to the System 80+ design were included in CESSAR-DC Chapter 20 and were evaluated by the staff. Based on these evaluations, the staff concludes that ABB-CE has adequately demonstrated compliance for the USIs and medium- and high-priority GSIs that are technically relevant to the System 80+ design as required by 10 CFR 52.47(a)(1)(iv). Some of these items involve combined operating license (COL) action items and will be the responsibility of the COL applicant. In some cases in this chapter, the staff also refers to the "owner/operator" of the plant because ABB-CE sometimes refers, in its discussions of the USIs and GSIs in this chapter, to the owner/operator instead of to the COL applicant for the plant or the procedures discussed should not be the responsibility of the COL applicant.

Compliance with 10 CFR 52.47(a)(1)(ii)

As stated above, 10 CFR 52.47(a)(1)(ii) requires an design certification application to demonstrate compliance with any technically relevant portions of the TMI Action Plan requirements in 10 CFR 50.34(f). ABB-CE addressed these requirements in CESSAR-DC Chapter 20 and these requirements are discussed in Section 20.6 of this chapter. Because the overlap between these TMI Action Plan items and those from NUREG-0933 (discussed in Section 20.4 of this report) all the relevant 50.34(f) TMI Action Plan items are listed in Section 20.6 in tabular form. This provides the issue designation and a reference to the appropriate issue in Section 20.4 of this chapter which contains the evaluation of the 50.34(f) TMI Action Plan item.

The staff concludes that ABB-CE has adequately demonstrated compliance of the System 80+ design for the technically relevant portions of 10 CFR 50.34(f).

Incorporation of Operating Experience

In a staff requirements memorandum (SRM) from the Commission, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to ensure that the design certification process preserves operating experience insights in the certified design. As discussed in Section 20.7 of this chapter, the staff concludes that ABB-CE has adequately considered operating experience identified in generic letters and bulletins issued by the Commission since the beginning of 1980 in the System 80+ design.

Resolution of Issues Relevant to the System 80+ Design

In CESSAR-DC Section 20.1, ABB-CE listed the issues in Supplement 15 of NUREG-0933 that it considered relevant to the System 80+ design. The section also provides ABB-CE's justification for considering an issue not relevant to the design. The resolutions of the issues that ABB-CE and the staff considered relevant are discussed in Sections 20.2 through 20.6 of this chapter.

These sections also address issues that ABB-CE did not consider relevant in Amendment U of CESSAR-DC and the staff does not consider relevant in terms of Supplement 15 of NUREG-0933. The staff evaluated these issues during the review of the System 80+ design since ABB-CE submitted CESSAR-DC in 1989, and decided to keep the evaluations in this chapter.

In Table 20.1, the staff lists the USIs and GSIs relevant to the System 80+ design, the sections in which these issues appear in this chapter, and the basis for the relevancy of each issue to the design. The relevancy of the issues fall into one of the following:

- the issue is required by 10 CFR 52.47(a)(1)(ii) or (iv) (i.e., 52.47).
- the issue was selected by ABB-CE as being relevant in CESSAR-DC Chapter 20 (i.e., ABB-CE).
- the staff decided to discuss the issue (i.e., staff).

In the latter case, ABB-CE originally stated the issue was relevant in an early amendment to Chapter 20 of CESSAR-DC and later concluded that the issue was not relevant to the System 80+ design. These issues and the staff evaluations are arranged in Table 20.1 in the order in which they appear in Sections 20.2 through 20.5 of this chapter.

20.2 Task Action Plan Items

With the exception of Issues A-48 and B-26, the task action plan items are evaluated against the System 80+ design in this section:

- for the design to comply with 10 CFR 52.47(a)(1)(iv) and 10 CFR 50.34(f)
- because ABB-CE stated in CESSAR-DC Table 20.1-1 that the task action plan item applied to the design

The staff also decided to include a discussion of Issues A-48 and B-26 for the System 80+ design.

Issue A-1: Water Hammer

Issue A-1, in NUREG-0933, addresses the issue of water hammer in fluid systems in nuclear power plants. Water hammer can be caused by a number of conditions, such as voiding in normally filled lines, condensation in lines, entrainment of water in steam-filled lines, or rapid valve actuation. Issue A-1 addresses these probable causes, as well as possible methods for minimizing the susceptibility of systems to water hammer through design and operational considerations. This issue was resolved with the publication of NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," Revision 1, dated March 1984, which contained evaluation results of water hammer events, as well as details of recommendations and measures for water hammer prevention and mitigation.

In CESSAR-DC Section 20.2.50, ABB-CE addresses the issue of water hammer through a combination of design, operational, and testing considerations, such as designing for the proper routing and sloping of lines, providing adequate drainage and venting to protect against water or

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design

Issue	Title of Issue and Section of this Chapter	Relevancy
	Section 20.1, Task Action Plan Items	
A-1	Water Hammer	52.47/CE
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	52.47/CE
A-4	ABB-CE Steam Generator Tube Integrity	52.47/CE
A-9	Anticipated Transient Without Scram (ATWSs)	52.47/CE
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Supports	52.47/CE
A-13	Snubber Operability Assurance	52.47/CE
A-17	Systems Interactions in Nuclear Power Plants	52.47/CE
A-24	Qualification of Class 1E Safety-Related Equipment	52.47/CE
A-25	Non-Safety Loads on Class 1E Safety-Related Equipment	52.47/CE
A-26	Reactor Vessel Pressure Transient Protection	52.47/CE
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Sabotage	52.47/CE
A-30	Adequacy of Safety-related DC Power Supplies	CE
A-31	RHR Shutdown Requirements	52.47/CE
A-35	Adequacy of Offsite Power Systems	52.47/CE
A-36	Control of Heavy Loads Near Spent Fuel	52.47/CE
A-40	Seismic Design Criteria Short-term Program	52.47/CE
A-43	Containment Emergency Sump Performance	52.47/CE
A-44	Station Blackout	52.47/CE
A-45	Shutdown Decay Heat Removal Requirements	CE
A-47	Safety Implications of Control Systems	52.47/CE
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Staff
A-49	Pressurized Thermal Shock	52.47/CE
B-5	Ductibility of Two-Way Slabs and Shells, and Buckling Behavior of Steel Containments	52.47/CE
B-17	Criteria for Safety-Related Operator Actions	52.47/CE
B-26	Structural Integrity of Containment Penetrations	Staff
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for ESF Systems and Normal Ventilation Systems	52.47/CE
B-53	Load Break Switch	CE
B-56	Diesel Reliability	52.47/CE
B-60	Loose-Parts Monitoring Systems	CE
B-61	Allowable ECCS Equipment Outage Periods	52.47/CE
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	52.47/CE
B-66	Control Room Infiltration Measurements	52.47/CE

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design (continued)

Issue	Title of Issue and Section of Chapter	Relevancy
	Section 20.1, Task Action Plan Items	
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	52.47/CE
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	CE
C-4	Statistical Methods for ECCS Analysis	CE
C-5	Decay Heat Update	CE
C-10	Effective Operation of Containment Sprays in a LOCA	52.47/CE
	Primary System Vibration Assessment	
C-12	Interim Acceptance Criteria for Solidification Agents for Radioactive	CE
C-17	Solid Wastes	52.47
	Section 20.3, New Generic Issues	
3	Setpoint Drift in Instrumentation	CE
14	PWR Pipe Cracks	CE
15	Radiation Effects on Reactor Vessel Supports	52.47/CE
22	Inadvertent Boron Dilution Events	CE
23	Reactor Coolant Pump Seal Failures	52.47/CE
24	Automatic ECCS Switchover to Recirculation	52.47/CE
29	Bolting Degradation or Failure in Nuclear Power Plants	52.47/CE
36	Loss of Service Water	CE
43	Reliability of Air Systems	CE
45	Inoperability of Instrumentation Due to Extreme Cold Weather	52.47/CE
48	LCO [Limiting Condition for Operation] for Class 1E Vital Instrument Buses in Operating Reactors	CE
49	Interlocks and LCOs for Class 1E Tie Breakers	CE
51	Improving the Reliability of Open-Cycle Service Water Systems	52.47/CE
57	Effects of Fire Protection Systems Actuation on Safety-Related Equipment	52.47/CE
64	Identification of Protection System Instrument-Sensing Lines	CE
66	Steam Generator Requirements	CE
67.3.3	Improved Accident Monitoring	52.47/CE
70	PORV and Block Valve Reliability	52.47/CE
75	Generic Implications of ATWS Events at Salem Nuclear Plant	52.47/CE
78	Monitoring Fatigue Transient Limits for the Reactor Coolant System	52.47/CE
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	CE
82	Beyond-Design-Basis Accidents in Spent Fuel Pools	52.47/CE
83	Control Room Habitability	52.47/CE
84	ABB-CE PORV	52.47

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design (continued)

Issue	Title of Issue and Section of this Chapter	Relevancy
	Section 20.3, New Generic Issues	
87	Failure of HPCI Steam Line Without Isolation	52.47/CE
93	Steam Binding of Auxiliary Feedwater Pumps	52.47/CE
94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors	52.47/CE
99	RCS/RHR Suction Line Valve Interlock on PWRs	52.47/CE
103	Design for Probable Maximum Precipitation	52.47/CE
105	Interfacing System LOCA at LWRs	CE
106	Piping and Use of Combustible Gases in Vital Areas	52.47/CE
113	Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	52.47/CE
118	Tendon Anchorage Failure	52.47/CE
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	CE
119.2	Piping Damping Values	CE
119.3	Decoupling the OBE from the SSE	CE
119.5	Leak Detection Requirements	CE
120	On-Line Testability of Protection Systems	52.47/CE
121	Hydrogen Control for Large, Dry PWR Containments	52.47/CE
122.2	Initiating Feed and Bleed	CE
124	Auxiliary Feedwater System Reliability	52.47/CE
125.I.3	Safety Parameter Display System Availability	CE
125.II.7	Reevaluate Provisions To Automatically Isolate Feedwater from Steam Generator During a Line Break	CE
128	Electric Power Reliability	52.47/CE
130	Essential Service Water Pump Failures at Multiplant Sites	52.47/CE
135	Steam Generator and Steamline Overfill	52.47/CE
142	Leakage Through Electrical Isolators in Instrumentation Circuits	52.47/CE
143	Availability of Chilled Water Systems and Room Cooling	52.47/CE
153	Loss of Essential Service Water in LWRs	52.47/CE
155.1	More Realistic Source-Term Assumptions	52.47/CE
	Section 20.4, Three Mile Island Action Plan items	
I.A.1.4	Long-Term Upgrade of Operating Personnel and Staffing	52.47
I.A.4.1(2)	Interim Changes in Training Simulators	52.47
I.A.4.2	Long-Term Training Simulator Upgrade	52.47
I.C.1	Guidance for Evaluation and Development of Procedures for Transients and Accidents	CE
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	Staff
I.C.9	Long-Term Program for Upgrading Procedures	52.47/CE
I.D.1	Control Room Design Reviews	52.47/CE
I.D.2	Plant Safety Parameter Display Console	52.47/CE

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design (continued)

Issue	Title and Section of this Chapter	Relevancy
	Section 20.4, Three Mile Island Action Plan items	
I.D.3	Safety System Status Monitoring	52.47/CE
I.D.4	Control Room Design Standard	CE
I.D.5(1)	Control Room Design: Improved Instrumentation Research - Alarms and Displays	CE
I.D.5(2)	Control Room Design: Improved Instrumentation Research - Plant Status and Postaccident Monitoring	52.47/CE
I.D.5(3)	Control Room Design: On-Line Reactor Surveillance Systems	52.47/CE
I.D.5(4)	Improved Control Room Instrumentation: Process Monitoring Instrumentation	CE
I.F.1	Expanded Quality Assurance	52.47/CE
I.F.2	Development of More Detailed QA Criteria	52.47/CE
I.G.2	Scope of Test Program	52.47/CE
II.B.1	Reactor Coolant System Vents	52.47/CE
II.B.2	Plant Shielding to Provide Postaccident Access to Vital Areas	52.47/CE
II.B.3	Postaccident Sampling Capability	52.47/CE
II.B.8	Rulemaking Proceedings on Degraded Core Accidents Description	52.47/CE
II.C.4	Reliability Engineering	CE
II.D.1	Performance Testing of PWR Safety and Relief Valves	52.47/CE
II.D.3	Coolant System Valves: Valve Position Indication	52.47/CE
II.E.1.1	Auxiliary Feedwater System Evaluation	52.47/CE
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	52.47/CE
II.E.1.3	Updated Standard Review Plan and Development of Regulatory Guides	52.47
II.E.2.2	Research on Small LOCAs and Anomalous Transients	Staff
II.E.3.1	Pressurizer Heater Power Supply	52.47/CE
II.E.4.1	Dedicated Hydrogen Penetrations	52.47/CE
II.E.4.2	Containment Isolation Dependability	52.47/CE
II.E.4.4	Purging	52.47/CE
II.E.6.1	In Situ Valve Testing	52.47/CE
II.F.1	Additional Accident Monitoring Instrumentation	52.47/CE
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	52.47/CE
II.F.3	Instrumentation for Monitoring Accident Conditions	52.47/CE
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	52.47/CE
II.J.3.1	Organization and Staffing to Oversee Design and Construction	52.47/CE
II.J.4.1	Revise Deficiency reporting requirements	52.47
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation In Transients and Accidents	CE

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design (continued)

Issue	Title and Section of this Chapter	Relevancy
	Section 20.4, Three Mile Island Action Plan items	
II.K.1(4d)	Review Operating Procedures and Training To Ensure That Operators Are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions	CE
II.K.1(5)	Safety-Related Valve Position Description	52.47/CE
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	CE
II.K.1(9)	Review Procedures To Ensure That Radioactive Liquids and Gases Are Not Transferred Out of Containment	CE
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	52.47/CE
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementing of All Bulletin Items	52.47
II.K.1(14)	Review Operating Modes and Procedures To Deal With Significant Amounts of Hydrogen	CE
II.K.1(15)	For Facilities With Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication With the Control Room To Operate AFW	CE
II.K.1(16)	Implemented Procedures That Identify Pressurizer PORV "Open" Indications and That Direct Operator to Close Valve Manually at "Reset" Setpoint	CE
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	52.47/CE
II.K.1(25)	Develop Operator Action Guidelines Position and Resolution	52.47/CE
II.K.1(26)	Revise Emergency Procedures and Train Reactor Operators and Senior Reactor Operators	52.47/CE
II.K.1(27)	Provide Analysis and Develop Guidelines and Procedures for Inadequate Core Cooling	52.47/CE
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	52.47/CE
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	52.47/CE
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident	CE
II.K.3(6)	Instruments To Verify Natural Circulation	CE
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	CE
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	CE
II.K.3(30)	Revise Small-Break LOCA Methods To Show Compliance With 10 CFR Part 50, Appendix K	CE
II.K.3(31)	Plant-Specific Calculations To Show Compliance With 10 CFR 50.46	CE
II.K.3(55)	Operator Monitoring of Control Board	CE
III.A.1.2	Upgrade Licensee Emergency Support Facilities	52.47/CE

Table 20.1 USIs/GSIs in NUREG-0933 (Supplement 15) relevant to the System 80+ Design (continued)

Issue	Title and Section of this Chapter	Relevancy
	Section 20.4, Three Mile Island Action Plan items	
III.A.3.3	Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Communication Systems	52.47
III.D.1.1	Primary Coolant Sources Outside the Containment	52.47/CE
III.D.3.3	In-Plant Radiation Monitoring	52.47/CE
III.D.3.4	Control Room Habitability	52.47/CE
	Section 20.5, Human Factors Issues	
HF1.1	Shift Staffing	52.47
HF4.4	Guidelines for Upgrading Other Procedures	52.47
HF5.1	Local Control Station	52.47/CE
HF5.2	Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation	52.47/CE

NOTES:

* 52.47: The resolution of the issue is required by 10 CFR 52.47(a)(1)(ii) and (iv).
CE: Although not required by 52.47, ABB-CE submitted an evaluation in CESSAR-DC Chapter.
Staff: The staff provided a resolution for the issue although ABB-CE did not provide an evaluation in CESSAR-DC Chapter 20.

steam entrainment, and consideration in the design analysis of dynamic loads resulting from water hammer. In the draft safety evaluation report (DSER) on the System 80+ design, the staff concluded that, although the actionsproposed by ABB-CE to eliminate or reduce the occurrences of water hammer were acceptable, ABB-CE had not submitted (1) the proposed guidelines for the owner/operator for hot functional testing, operation, and maintenance and (2) the methodology for consideration of dynamic loads on piping systems resulting from water hammer. These two items were identified as DSER Open Item 20.1-1.

In response to DSER Open Item 20.1-1, ABB-CE provided in Amendment U to CESSAR-DC Section 20.2.50 that general guidelines and associated references for use by the COL applicant in preparing plant operating and maintenance procedures to minimize the potential for water hammer. Guidelines will be provided to the COL applicant for hot functional testing, as well as operating and maintenance procedures that require proper precautions to minimize the potential for water hammer. The staff concludes that these guidelines are consistent with the staff's recommendations in NUREG-0927 and are acceptable.

In CESSAR-DC Section 1.4.5.2, Appendix 3.9A, ABB-CE indicates that piping systems are evaluated for water and steam hammer loading using time history dynamic solutions with the force-time histories as input loading. Water and steam hammer force-time histories are usually developed using method-of-characteristics or applicable computer codes. This meets the guideline of Standard Review Plan (SRP) Section 3.9.3, which states that the potential for water and steam hammer events should be given proper consideration in service loading combinations. In addition, ABB-CE commits to implement the guidance identified in SRP Sections 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, and 10.4.7 (including Branch Technical Position (BTP) ASB 10-2) for preventing damage to various safety-related systems from water hammers.

The staff, therefore, concludes that ABB-CE's proposed actions to eliminate or reduce the occurrences of water hammer and the potential for water hammer are acceptable. On this basis, DSER Open Item 20.1-1 and Issue A-1 are resolved for the System 80+ design.

Issue A-2: Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

In a postulated event of reactor coolant pipe rupture at the vessel nozzle, asymmetric loss-of-coolant accident (LOCA) loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and forces on the vessel associated with transient differential pressures in the reactor cavity. This was designated Issue A-2 in NUREG-0933.

This issue was resolved in January 1981, with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems." To resolve A-2, the asymmetric loads on the reactor vessel, internals, primary coolant loop, and components should not exceed the limits imposed by the applicable codes and standards. The staff also issued Generic Letter (GL) 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," on February 1, 1984, to permit the application of leak-before-break (LBB) technology to eliminate the postulated pipe rupture from the design basis. Subsequently, the staff revised General Design Criterion (GDC) 4 to permit the application of LBB.

In CESSAR-DC Section 3.6.3, ABB-CE proposes to use LBB methodology to eliminate the postulated pipe rupture from the design basis. The staff evaluated the LBB methodology in Section 3.6.3 of this report and approved its application to the System 80+ design. Where the LBB approach cannot be applied, ABB-CE states that the pipe break locations and resulting dynamic effects are determined. Each postulated pipe rupture is considered separately as a single postulated initiating event.

In the DSER, the staff stated if ABB-CE could not obtain staff approval for the LBB approach, it would have to submit details of its analysis on assessing the effects of the asymmetric blowdown loads. This was designated as DSER Open Item 20.1-2. On the basis that the staff has approved the application of LBB for the System 80+ design, this DSER Open Item 20.1-2 is resolved.

On the basis of the above, ABB-CE's proposal for resolving Issue A-2 is adequate to address asymmetric blowdown loads and, thus, is acceptable in resolving Issue A-2 for the System 80+ design.

Issue A-4: ABB-CE Steam Generator (SG) Tube Integrity

Staff concerns related to steam generator (SG) tube degradation stem from the fact that the SG tubes are a part of the reactor coolant system (RCS) boundary, and that tube ruptures allow primary coolant into the secondary system where its isolation from the environment is not fully ensured. In 1978, Issues A-3, A-4, and A-5 were established to evaluate the safety significance of tube degradation in Westinghouse, ABB-CE, and Babcock and Wilcox SGs, respectively. These studies were later combined into one effort because of the similarity of many problems among the pressurized water-reactor (PWR) vendors.

This issue was resolved and no new requirements were established (U.S. NRC, "Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," SECY-88-272, September 27, 1988; and "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," NUREG-0844, September 1988). However, the staff issued GL 85-02, "Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity," dated April 17, 1985, to provide recommended actions from NUREG-0844. After reviewing responses to GL 85-02, the staff concluded that the large majority of licensees and applicants are following programs, practices, and procedures that are partially to fully consistent with, or equivalent to, the recommendations discussed in GL 85-02.

ABB-CE states in CESSAR-DC Section 20.2.52 that it will comply with the following recommendations in GL 85-02, to the extent that they are applicable to the System 80+ design:

- prevention and detection of loose parts.
- SG tube in-service inspection (ISI).
- secondary water chemistry and impurity control.
- primary-to-secondary coolant leakage limit.
- use of Nitrogen-16 and area radiation monitors.
- primary coolant iodine activity limit.
- safety injection (SI) signal reset logic.

ABB-CE also states that Inconel 690 will be used for SG tubes to provide increased resistance to corrosion.

The staff finds that ABB-CE's proposed resolution to Issue A-4 for the System 80+ design is acceptable; however, ABB-CE will be subject to the staff's proposed applicable regulations on SG tube integrity which are addressed in Section 15.3.9 of this report.

The initial staff reviews identified an unresolved issue regarding secondary water chemistry guidelines. This issue was designated as DSER Open Item 5.4.2-5. As stated in Section 5.4.2 of this report, the secondary water chemistry guidelines contained in the CESSAR-DC now conform to the recently published Electric Power Research Institute (EPRI) guidelines for makeup water to SGs. Therefore, DSER Open Item 5.4.2-5 is resolved.

As discussed in Sections 5.4.2 and 15.3.9 of this report, ABB-CE specifies that development of the SG tube ISI program is the responsibility of the COL applicant. The program is plant specific and will be reviewed by the staff individually for each license application referencing the System 80+ design certification. Therefore, submittal of the ISI program is designated as COL Action Item 5.F-1.

See Issues 66 and 135 (Section 20.3 of this chapter) for additional evaluations of SG issues.

Issue A-9: Anticipated Transient Without Scram

Issue A-9, in NUREG-0933, addressed the issue of ensuring that the reactor can attain safe shutdown after incurring an anticipated transient with a failure of the reactor trip system (RTS). An anticipated transient without scram (ATWS) is an expected operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power (LOOP) to the reactor) that is accompanied by a failure of the RTS to shut down the reactor.

The acceptance criterion for the resolution of Issue A-9 is that the reactor must be capable of reaching a safe-shutdown condition as identified in 10 CFR 50.62, after incurring an anticipated transient and an RTS failure:

- To comply with the mitigation requirement of 10 CFR 50.62(c)(1), plant equipment must automatically initiate emergency feedwater (EFW) and turbine trip under conditions indicative of an ATWS. This equipment must function reliably and must be diverse and independent from the RTS.

- To comply with the prevention requirement of 10 CFR 50.62 (c)(2), the plant must have a scram system that is diverse and independent from the existing RTS.

In CESSAR-DC Section 20.2.53, ABB-CE states that the System 80+ design contains safety-grade and control-grade systems that are designed to protect the plant and mitigate the consequences of design-basis events (DBEs). These systems have the following design features:

- The plant protection system (PPS) consists of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS). The PPS is designed with both redundancy and diversity to maximize the ability to mitigate transients. However, should an ATWS occur, the System 80+ design includes an alternate protection system (APS) for mitigation.

- The APS augments the RPS to address 10 CFR 50.62 requirements for the reduction in risk of ATWS and for the use of ATWS mitigating systems actuation circuitry (AMSAC).

  ABB-CE states that the APS design includes an alternate reactor trip signal (ARTS) and an alternate feedwater actuation signal (AFAS) that are separate and diverse from the PPS. The APS equipment provides diverse and independent mechanisms to reduce the possibility of an ATWS and to offer additional assurance that an ATWS event could be mitigated.

- The ARTS will initiate a reactor trip when the pressurizer pressure exceeds a predetermined value. Turbine trip signals can also initiate the ARTS if the reactor power cutback system (RPCS) is out of service. The ARTS turbine trip input is manually enabled from the main control panel.

  ABB-CE states that the ARTS circuitry is diverse and independent from that of the RPS. The ARTS design uses a 2-out-of-2 logic to open the motor-generator output contractors, thus, removing motive power to the reactor trip switchgear system (RTSS).

- The AFAS will start EFW to a SG when the water in that SG decreases below a predetermined level. Its circuitry is diverse from that of the RPS. The EFW pumps and valves are actuated by sending isolated AFAS signals to the ESFAS.

Based upon the initial description of the System 80+ ATWS design submitted by ABB-CE, the staff concluded in Section 7.7 of the DSER that the ATWS implementation was not acceptable. Subsequently, ABB-CE submitted a revised description of the ATWS design that the staff found acceptable, thereby resolving DSER Open Items 7.7.1.12-1 and 7.7.1.12-2. The staff now finds that the ATWS is designed to perform its function independent of the RTS. This is discussed in Section 7.7.1.12 of this report.

On this basis, Issue A-9 is resolved for the System 80+ design.

Issue A-12: Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

During the course of the licensing action for North Anna, Units 1 and 2, a number of questions were raised about the potential for lamellar tearing and low-fracture toughness of the SG and reactor coolant pump (RCP) support materials for these facilities. Concerns regarding the supports at North Anna were applicable to all PWRs. This was designated as Issue A-12 in NUREG-0933.

This issue was resolved and no new requirements were established (U.S. NRC, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," NUREG-0577, Revision 1, October 1983). However, the staff recommended developing guidance for new plants based on the fracture toughness requirements of Subsection NF of Section III of the American Society of Mechanical Engineers (ASME) Code.

ABB-CE states in CESSAR-DC Section 20.2.54 that the major RCS component supports, including those for the SGs and the RCPs, will comply with the requirements in ASME Code, Section III, Subsection NF. Thus, the ABB-CE proposal adequately addresses the structural integrity of SG and RCP supports and Issue A-12 is resolved for the System 80+ design.

Issue A-13: Snubber Operability Assurance

Snubbers are primarily used as seismic and pipe whip restraints at nuclear power plants. They function as rigid supports for restraining the motion of attached systems or components under such rapidly applied load conditions as earthquakes, pipe breaks, and severe hydraulic transients, while allowing free expansion of the systems and components during various operating conditions. Issue A-13 in NUREG-0933 addressed the concern of a substantial number of snubber malfunctions, the most frequent of which were (1) seal leakage in hydraulic snubbers and (2) high rejection rate during functional testing of snubbers. This issue has been resolved and new requirements were established with the revision of SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," in 1981.

By request for additional information (RAI) Q210.65, listed in Appendix B of this report, the staff asked ABB-CE to submit information identified in SRP Section 3.9.3 (Rev. 1), Subsection II.3.6(7), regarding safety-related components that use snubbers. This information was to include the following data:

- Identify the systems and components in those systems which utilize snubbers.

- Specify the number of snubbers utilized in each system and on components in that system.

- Specify the type(s) of snubber (hydraulic or mechanical) and identify the corresponding supplier.

- Specify whether or not the snubber was constructed to the rules of ASME Code, Section III, Subsection NF.

- State whether or not the snubber is used as a shock, vibration, or dual- purpose snubber.

- For snubbers identified as either dual purpose or vibration arrester type, indicate whether or not both snubber and components were evaluated for fatigue strength.

In response to staff RAI Q210.65, ABB-CE stated that a listing of all safety-related components that use snubbers (including the requested detailed information) requires detailed plant arrangements, piping layouts, and piping designs. Because detailed piping system design and layout and plant arrangements are not required at the design certification stage, the staff requested, by DSER COL Action Item 20.2-9, that the COL applicant shall submit a list of all safety-related components that use snubbers per SRP Section 3.9.3. ABB-CE states in CESSAR-DC Section 3.9.3.4 and lists in CESSAR-DC Table 1.10-1 that the COL applicant will submit the requisite list to the NRC staff. This is designated COL Action Item 3.9.3.4-1, which is discussed in Section 3.9.3.4 of this report. On this basis, DSER COL Action Item 20.2-9 is resolved.

In CESSAR-DC Sections 3.9.3 and 20.2.55, ABB-CE also provides the general design and operability assurance acceptance criteria proposed for snubbers including large-bore hydraulic snubbers (LBHSs). The staff concludes that these criteria are acceptable and meet SRP Section 3.9.3.

Therefore, Issue A-13 is resolved for the System 80+ design.

Issue A-17: Systems Interactions in Nuclear Power Plants

Issue A-17, in NUREG-0933, addressed the concerns regarding adverse systems interactions (ASIs) in nuclear power plants. Depending on how they propagate, ASIs can be classified as functionally coupled, spatially coupled, and induced-human-intervention coupled. As discussed in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," dated August 1989, and GL 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989, Issue A-17 concerns ASIs caused by water intrusion, internal flooding, seismic events, and pipe ruptures.

A nuclear power plant comprises numerous structures, systems, and components (SSCs) that are designed, analyzed, and constructed using many different engineering disciplines. The degree of functional and physical integration of these SSCs into any single power plant may vary considerably. Concerns have been raised about the adequacy of this functional and physical integration and coordination process. The Issue A-17 program was initiated to integrate the areas of systems interactions and consider viable alternatives for regulatory requirements to ensure that the ASIs have been or will be minimized in operating plants and new plants. Within the framework of the program, the staff requested, as stated in NUREG-0933, that plant designers consider the operating experience discussed in GL 89-18 and use the probabilistic risk assessment (PRA) required for future plants to identify the vulnerability and reduce ASIs.

In responding to staff RAI Q440.127(1), listed in Appendix B of this report, ABB-CE stated that System 80+ is designed to prevent ASIs resulting from water intrusion, internal floods, seismic events, and pipe ruptures and gave examples of these design features. In the resolution to Issue A-17 included in CESSAR-DC Section 20.2.56, ABB-CE states that the System 80+ design was evaluated for its vulnerability to ASIs identified from previous designs, and operating experiences reported in licensee event reports (LERs) and NRC information notices. ABB-CE evaluated each of the interaction incidents resulting from water intrusion referenced in NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants: Technical Findings Related to Unresolved Safety Issue A-17," dated May 1989, to identify the features of the System 80+ design that should ensure prevention of a similar ASI. In addition, the System 80+ PRA covers functionally coupled ASIs.

At the time the DSER was issued, ABB-CE was scheduled to revise the System 80+ PRA. The revision included plans to qualitatively assess potential fire and flood risk in order to partially address spatially coupled ASIs. Spatially coupled ASIs were also addressed, in part, by the seismic PRA. ABB-CE committed to evaluate induced-human-intervention-coupled ASIs in parallel with the System 80+ PRA revision and to submit an inspections, tests, analyses, and acceptance criteria (ITAAC) program acceptable to the NRC for ASI risk reduction. The staff stated in the DSER that it would evaluate the PRA submittals and ITAAC program provided by ABB-CE and include an evaluation of Issue A-17 resolution results in this report. In addition, ABB-CE was to submit the requirements for conducting walkdowns at "as built" plants to identify any spatial interactions. These actions were designated as DSER Open Item 20.1-3.

Since the DSER was issued, ABB-CE has updated the System 80+ PRA in CESSAR-DC Chapter 19 and the staff has concluded that the issues relating to the closure of DSER Open Item 20.1-3 have been acceptably addressed. Nevertheless, spatially coupled ASIs and walkdowns of the as-built plant are issues that will be addressed by the COL applicant. This is part of COL Action Item 19.8 which is discussed in Section 19.1 of this report.

In addition, the staff has reviewed the System 80+ ITAAC program and concluded that there are no open items relating to the resolution of Issue A-17.

Therefore, Issue A-17 is resolved for the System 80+ design.

Issue A-24: Qualification of Class 1E Safety-Related Equipment

Construction permit (CP) applicants for which safety evaluation reports (SERs) were issued after July 1, 1974, were required by the NRC to qualify all safety-related equipment to Institute of Electrical and Electronics Engineers (IEEE)-323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time this standard was originated, the industry developed methods that were used to qualify equipment in accordance with the standard. Some of these methods had not been resolved to the satisfaction of the NRC. To assess the adequacy of the equipment qualification methods and acceptance criteria used by nuclear steam supply system (NSSS) and balance-of-plant (BOP) vendors, the NRC determined that a generic approach was required. This was designated as Issue A-24 in NUREG-0933 and was resolved with the publication of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated July 1981.

This issue on environmental design and qualification is discussed in CESSAR-DC Sections 3.11 and 20.2.57. The Class 1E electrical equipment (including pump and valve motors and electrical accessories) of the System 80+ design is environmentally qualified by the methods documented in the NRC-approved report CENPD-255-A, "Class 1E Qualification" (Rev. 3, October 1985). ABB-CE states that the methods in CENPD-255-A are in accordance with the guidance of IEEE 323-1974, NUREG-0588, Regulatory Guide (RG) 1.89 ("Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants", Rev. 1), and the generic requirements of 10 CFR 50.49, as described in CESSAR-DC Section 3.11. The staff approved CENPD-255-A in its letter to ABB-CE dated August 13, 1985.

Equipment supplied by ABB-CE, and either required to mitigate the consequences of a design-basis accident (DBA) or attain a safe shutdown of the reactor is tabulated in CESSAR-DC Appendix 3.11B. This tabulation indicates both the location of the equipment and, in part by a cross-reference to Appendix 3.11A, typical normal and accident environments in that location, including integrated radiation doses.

Typical environmental conditions (temperature, pressure, humidity, integrated radiation dose, and exposure to chemicals) are given in CESSAR-DC Appendix 3.11A for the 60-year design lifetime. Conditions are tabulated for normal operation inside and outside of the containment, and for LOCA and main steamline break (MSLB) inside the containment.

Environmental qualification tests and analyses are addressed in CESSAR-DC Section 3.11.2. The equipment listed in Appendix 3.11B is stated by ABB-CE to be environmentally qualified for 60 years exposure to normal operating conditions (not required by NRC) and then to remain functional in the environmental conditions expected at the equipment location during and after the limiting DBA. The environmental conditions are tabulated in Appendix 3.11A. Qualification tests and analyses of electrical equipment for the effects of aging, radiation, temperature, humidity, chemical spray, submergence, and power supply variation, as applicable, are performed, and the results are documented in accordance with CENPD-2-55-A (Rev. 3).

With the exception of pump motors and valve motor operators, which were addressed in CESSAR-DC Section 3.9.2.2, dynamic and seismic qualification testing and analysis of the electrical equipment listed in CESSAR-DC Appendix 3.11B are addressed in CESSAR-DC Section 3.10. The tests and analyses are performed in accordance with IEEE 344-1987, which is endorsed by RG 1.100. However, when the DSER was issued, ABB-CE was revising CESSAR-DC Section 3.10. Therefore, the staff was unable complete its review of the ABB-CE submittal. This was designated as DSER Open Item 3.10-1 which has been resolved as discussed in Section 3.10 of this report.

On the basis of the staff's review, which is discussed in Section 3.11 of this report, the staff concludes that ABB-CE's approach to environmental qualification of Class 1E equipment is in compliance with 10 CFR 50.49 and Issue A-24 is resolved for the System 80+ design.

In the DSER, the staff stated that ABB-CE's approach to resolve this issue was acceptable with the exception that ABB-CE stated in CESSAR-DC Amendment I, that CENPD-255-A (Rev. 3) was in accordance with IEEE-323-1983, instead of IEEE 323-1974. The exception is that the staff had approved CENPD-255-A (Rev. 3) in part because it was in accordance with IEEE 323-1974 and the staff has not accepted IEEE 323-1983. This exception was designated DSER Open Item 20.1-4 and the staff requested ABB-CE to confirm that report CENPD-255-A was in accordance with IEEE 323-1974. ABB-CE now states in CESSAR-DC Section 20.2.57 and Appendix 19A that CENPD-255-A (Rev. 3) is in accordance with IEEE 323-1974, instead of IEEE 323-1983. This resolved DSER Open Item 20.1-4.

Issue A-25: Non-Safety Loads on Class 1E Safety-Related-Equipment

Issue A-25, in NUREG-0933, addressed a review of whether non-safety-related loads should also be allowed to share the Class 1E power sources. The Class 1E power sources provide the electric power for the plant systems that are essential to reactor shutdown, containment isolation, reactor core cooling, containment heat removal, and preventing significant release of radioactive material to the environment. As discussed in NUREG-0933, this issue was resolved in Revision 2 to RG 1.75, "Physical Independence of Electric Systems."

Issues A-25 is discussed in Section 8.3.1.8 of this report and, based on the staff's conclusions in this section, Issue A-25 is resolved for the System 80+ design.

Issue A-26: Reactor Vessel Pressure Transient Protection

Since 1972, there have been, since 1972, many reported pressure transients which have exceeded the pressure-temperature limits specified in technical specifications (TSs) for PWRs. The majority of these events occurred at relatively low reactor vessel temperatures at which the material has less toughness and is more susceptible to failure through brittle fracture. This is Issue A-26 in NUREG-0933 which was resolved with the issuance of SRP Section 5.2.2, "Overpressure Protection." Applicants for CPs and operating licenses were requested to design an overpressure protection system for light-water reactors (LWRs) following the guidance provided in SRP Section 5.2.2.

Overpressure protection for the System 80+ design is described in CESSAR-DC Sections 5.2.2, 5.4.10, 5.4.13 and Appendix 5A in accordance with SRP Section 5.2.2. Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by pressurizer safety valves, SG safety valves, and relief valves of the shutdown cooling system (SCS), in combination with the action of the RPS. The combination of these features provides overpressure protection as required by GDC 15, Section III of the ASME Code, and Appendix G of 10 CFR Part 50. These measures ensure RCPB overpressure protection for power operation and low temperature (startup and shutdown) operation.

The staff evaluated the overpressure protection system for the System 80+ design in Sections 5.2.2.1 and 5.2.2.2 of this report and determined that the system conforms to the requirements of GDC 15 and SRP Section 5.2.2. Therefore, Issue A-26 is resolved for the System 80+ design

In the DSER, the staff did not address this issue. The staff stated that it would address this issue in this report and designated this as DSER Open Item 20.1-5. Based on the above discussion, DSER Open Item 20.1-05 is resolved.

Issue A-29: Nuclear Power Plant Design for Reduction of Vulnerability to Sabotage

Issue A-29, in NUREG-0933, addressed alternatives to the basic design of nuclear power plants with the emphasis primarily on reducing the vulnerability of reactors to radiological sabotage. In the past, reduction in the vulnerability of reactors to such sabotage has been treated as a plant physical security function and not as a plant design requirement.

This issue is addressed in Section 13.6 of this report and, based on the staff's conclusions in this section, Issue A-29 is resolved for the System 80+ design.

Issue A-30: Adequacy of Safety-Related DC Power Supplies

Issue A-30, in NUREG-0933, addressed the adequacy of the safety-related dc power in operating plants. The dc power system in a nuclear power plant provides control and motive power to valves, instrumentation, emergency diesel generators (EDGs), and other components during normal and accident conditions. Assurance of dc supply reliability would require that the batteries and other systems elements remain ready for full operation and there is independence of the dc redundant divisions. An aspect of potential significance of Issue A-30 is that failure of one division would generally cause a reactor scram that could result in a demand for dc power to remove decay heat and prevent core melt. As stated in NUREG-0933, this issue was integrated into the resolution of Issue 128. Issue 128 is discussed in Section 20.3 of this chapter.

The Class 1E dc power system for the System 80+ design is described in CESSAR-DC Sections 8.3.2 and 20.2.61. It comprises four independent and physically separated subsystems that supply instrumentation and control (I&C) channels A, B, C, and D. Each Class 1E dc subsystem (i.e., batteries, charger, switchgear, and distribution system) is physically separate and independent from its redundant counterparts and non-Class 1E dc systems. The System 80+ design provides dedicated non-Class 1E dc systems for the non-Class 1E loads.

In order to ensure the continued operability of the Class 1E dc power systems, sufficient local and control room indication and alarms will monitor the status of the batteries and battery chargers. The staff evaluated these systems in Section 8.3.2 of this report.

In the DSER, the staff found that ABB-CE had not adequately addressed the requirement for loss of a dc bus not leading to a reactor trip. The staff stated that this issue would be resolved as part of the resolution of the open items discussed in Sections 8.3.2 and 8.3.2.1 of the DSER. In CESSAR-DC Amendment U, ABB-CE illustrated that the System 80+ design ensures that failure or loss of any dc bus does not result in a plant transient and simultaneously cause the loss of single-failure protection in any safety-related system. The requirement for reducing the probability of a reactor trip in the event of a loss of a single safety-related bus is described in Section 8.3.2 of this report.

Therefore, Issue A-30 is resolved for the System 80+ design.

Issue A-31: Residual Heat Removal Shutdown Requirements

Issue A-31, in NUREG-0933, addressed the ability to transfer heat from the reactor to the environment after a shutdown, which is an important safety function. It was resolved in 1978 with the issuance of SRP Section 5.4.7, "Residual Heat Removal (RHR) System."

The safe shutdown of a nuclear power plant following an accident not related to a LOCA has typically been interpreted as achieving "hot standby" condition. The NRC has placed considerable emphasis on the hot-standby condition of a power plant in the event of an accident or other abnormal occurrence and, similarly, on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby-condition values. Even though it may generally be considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that certain events have occurred that required eventual cooldown or long-term cooling until the RCS is cold enough for personnel to inspect the problem and repair it.

As discussed in CESSAR-DC Sections 5.4.7 and 20.2.62, the long-term RHR system is defined as the SCS for the System 80+ design. The SCS has entry conditions of 3105 kPa (450 psia) in pressurizer pressure and 177 °C (350 °F) in reactor temperature.

The staff reviewed the SCS for the System 80+ design in accordance with SRP Section 5.4.7 and BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System." The staff's evaluation in Section 5.4.3 of this report addresses compliance of the SCS design with the requirements of each of the following areas:

- functional requirements
- SCS isolation
- SCS pressure relief
- SCS pump protection
- tests, operational procedures, and support systems

The staff concluded that the SCS design for the System 80+ plant meets the requirements of BTP RSB 5-1, as discussed in Section 5.4.3 of this report. Therefore, Issue A-31 is resolved for the System 80+ design.

Issue A-35: Adequacy of Offsite Power Systems

Issue A-35, in NUREG-0933, addressed the adequacy of existing testing requirements and the susceptibility of safety-related electric equipment to a sustained degraded voltage condition on the offsite power source and an interaction of the offsite and onsite power sources. This issue included the following concerns:

- reliability of the offsite power systems as the preferred source
- vulnerability of Class 1E equipment to sustain degraded voltages
- interactions between offsite and onsite power sources
- adequacy of testing the onsite power sources.

This issue is addressed by BTP PSB-1 "Adequacy of Station Electric Distribution System Voltages," Appendix A to SRP Section 8.3.1.

The staff evaluated the conformance of System 80+ design to BTP PSB-1 in Section 8.3.1.14 of this report and, based on the staff's conclusions in this section, Issue A-35 is resolved for the System 80+ design.

Issue A-36: Control of Heavy Loads Near Spent Fuel

At all nuclear plants, overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, radioactivity could be released to the environment. Such an occurrence would also have the potential for overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases to the environment could exceed the exposure guidelines of 10 CFR Part 100. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there was a need for a systematic review of requirements, facility designs, and TSs regarding the movement of heavy loads to assess safety margins and improve them where necessary. This was designated as Issue A-36 in NUREG-0933.

As given in CESSAR-DC Section 20.2.64, ABB-CE addresses these criteria for the System 80+ design as follows:

(1) The component (heavy load) handling procedure guidelines will require the COL applicant to establish the safe load path and perform special handling component inspections before a lift.

(2) The plant operating procedure guidelines will require appropriate operator training and crane inspections.

(3) The cask-handling crane is designed with mechanical stops and electrical interlocks to prevent its movement near the spent fuel pool after the pool contains irradiated fuel (see CESSAR-DC Section 9.1.4).

(4) The new-fuel-handling crane is designed with mechanical stops and electrical interlocks to restrict its motion between the new-fuel shipping container receipt area, the new-fuel inspection and storage areas, and the new-fuel elevator (see CESSAR-DC Section 9.1.4).

(5) The spent-fuel building has been arranged so that the spent fuel cask does not pass over critical components during its travels from the shipping vehicle to the cask laydown area (see CESSAR-DC Sections 9.1.4.1.3 and 9.1.4.3.1).

(6) The reactor vessel head lift rig and the reactor vessel internal component lift rigs are designed in

accordance with the acceptable (stress) factors of safety as discussed in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

(7) A drop of the reactor vessel head onto the reactor vessel has been analyzed as described in CESSAR-DC Section 9.1.4.3.3, and the results are acceptable. (This was discussed in Section 9.1.4.2 of this report.)

(8) The upper guide structure drop on the reactor vessel has been analyzed to demonstrate that this event is bounded by the result of the analysis of item 7 (above).

(9) The load handling system is designed in accordance with the relevant requirements of GDC 2, 4, 5, and 61 (see CESSAR-DC Sections 3.1 and 9.1.4).

Heavy load handling equipment is discussed in Section 9.1.4.2 of this report in which the staff concludes that the design of the heavy load handling portions of the fuel handling system conform to GDC 2, 4, 5, and 61 and to the guidelines in SRP Section 9.1.5. Such compliance with GDC 2, 4, 5, and 61, and the guidance of SRP Section 9.1.5 is acceptable for resolving this issue for the System 80+ design. Also, all DSER open and confirmatory items in Section 9.1 of this report related to fuel storage and handling, have been resolved. Issue A-36 is, therefore, resolved for the System 80+ design.

Issue A-40: Seismic Design Criteria Short-Term Program

Issue A-40, in NUREG-0933, addressed short-term improvements in seismic design criteria. The objectives of Issue A-40 were the following:

- investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites.
- investigate alternative approaches, where desirable.
- quantify the overall conservatism of the design sequence.
- modify the NRC criteria in the SRP, where justified.

To resolve this issue, the staff revised SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3 to address areas of vibratory ground motion; design time-history criteria; development of floor response criteria, damping values, and soil-structure interaction (SSI) uncertainties; and combination of modal responses. The revisions also addressed seismic analysis of the above-ground tanks and Category I buried piping. An acceptable resolution of Issue A-40 is that future nuclear power plants should be required to conform to the seismic design acceptance criteria and guidance of Revision 2 to SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3.

In CESSAR-DC Section 20.2.65, ABB-CE states that the System 80+ design complies with Revision 2 of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Also, all tanks required to function during and after a safe-shutdown earthquake (SSE) are designed for SSE loads in accordance with SRP Section 3.7.3.

In Sections 2.5.2, 3.7.1, 3.7.2, 3.7.3, 3.9, and 3.10 of this report, the staff details its evaluation of vibratory ground motion, seismic design parameters, seismic analyses of systems and subsystems, seismic results of the coupled RCS, and the methodology and results of the SSI analysis. On the basis of these evaluations, Issue A-40 is resolved for the System 80+ design.

Issue A-43: Containment Emergency Sump Performance

Issue A-43, in NUREG-0933, concerns the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump (or boiling-water reactor (BWR) RHR suction intake) must be initiated and maintained to prevent core melt following a postulated LOCA. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired, thereby seriously degrading long-term recirculation flow capability.

The technical concerns evaluated under Issue A-43 are as follows:

- sump hydraulic performance under post-LOCA conditions resulting from potential vortex formation and air ingestion, and subsequent pump failure
- possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling
- capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

The staff issued its proposed resolution of Issue A-43 for public comment on May 10, 1983. The public comment package included draft NUREG-0869 ("USI A-43 Regulatory Analysis," dated October 1985), the staff's technical findings report draft NUREG-0897 ("Containment Emergency Sump Performance," dated October 1985), proposed RG 1.82 ("Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Rev. 1), and proposed SRP Section 6.2.2 ("Containment Heat Removal Systems," Rev. 4). The public comments received and the staff's responses were published in Appendix A of NUREG-0869 (Rev. 1). On October 31, 1985, the staff presented the resolution of Issue A-43 to the Commission in SECY-85-349, "Resolution of Unresolved Safety Issue A-43, Containment Emergency Sump Performance."

In CESSAR-DC Section 20.2.66, ABB-CE states that in the System 80+ design engineered safety features (ESFs) are incorporated to mitigate DBEs, including a LOCA. Two principal systems used to mitigate the effects of a LOCA are the safety injection system (SIS) (see CESSAR-DC Section 6.3), and the CSS (see CESSAR-DC Section 6.5). These systems use an in-containment refueling water storage tank (IRWST) as their single source of water. The IRWST is toroidal in shape, uses the lower section of the spherical containment as its outer boundary, and is enclosed to prevent contamination and excess containment humidity.

Long-term return of spray water from upper-level elevations is not dependent on individual floor screens and piping. Major openings, such as hatches and stairwells, are also available to return water to the screened entrance to the holdup volume.

It is ABB-CE's position that the IRWST meets the intent of SRP Section 6.2.2 (Rev. 4) and RG 1.82 (Rev. 1) with respect to

- IRWST hydraulic performance
- evaluation of potential debris generation and associated effects (including types and quantities of insulation, and debris screen blockage)
- preservation of NPSH for the SIS and CSS pumps after a LOCA
- multiple pathways to the IRWST for containment spray and SI water introduced into the containment building in case one drain becomes fouled with debris

The IRWST has the advantage that during normal full-power operation it is possible to perform a full-flow test of the SI pumps and containment spray pumps while taking suction from the IRWST and discharging back to the IRWST via a recirculation line. Satisfactory hydraulic performance of the IRWST can be verified by testing at runout conditions on the pumps and minimum level in the IRWST.

The System 80+ IRWST design differs from conventional sump designs. The IRWST does not function as the containment sump; the holdup volume tank (HVT) serves this purpose. Water, from a reactor coolant break or from the initiation of containment sprays, accumulates in the HVT and overflows into the IRWST via a spillway. Vertical screens, capable of filtering debris greater than 3.8 cm (1.5 in.) in diameter, are at the entrance of the HVT to prevent large debris from entering the HVT and thus the IRWST. These vertical screens are more than 6 ft high and more than 40 ft long. The HVT is of sufficient volume to allow a significant settling of high density debris.

The fine debris that could be introduced into the IRWST is prevented from entering the SIS suction header piping by a debris screen. These screens are located at each end of the four wing walls. These wing wall assemblies extend from the IRWST floor to above the maximum IRWST water level. The wing wall screens have the capability to remove particles greater than 0.23 cm (0.09 in.) in diameter. The IRWST screen design is described in CESSAR-DC Appendix 19.8A, Section 2.9.

Section 6.8.2.2.1 of CESSAR-DC requires that the COL applicant submit an analysis, consistent with RG 1.82, of the suction inlet screen area based on the insulation type and quantity in the containment. This analysis must show that the System 80+ screen are is at least three times over that indicated by RG 1.82. The staff finds this commitment sufficient to meet the staff's current position that emergency core cooling system (ECCS) suction strainers be sized in accordance with RG 1.82, Revision 1, but with a factor of three sizing margin.

Other design features have been incorporated to reduce the potential for a decrease in ECCS suction efficiency. To minimize the potential for corrosion products, IRWST surfaces that are in direct contact with borated water are lined with stainless steel and IRWST water can be cleaned by the chemical and volume control system (CVCS). Each of the four SIS pumps have separate IRWST suction lines and each of the two CSS pumps takes suction from one of these four lines. Finally, in response to staff RAI 440.166, listed in Appendix B of this report, ABB-CE stated that permanent cage-type vortex suppressor will be placed over each ECCS suction inlet to suppress vortices and eliminate air ingestion.

ABB-CE states in CESSAR-DC Section 13.5.2 that the containment must be cleaned of sand, maintenance debris, and other particulate materials prior to startup from a refueling outage to avoid excessive fouling and plugging of the screens near the IRWST suction inlets during an accident.

Several significant events have occurred at operating plants, including the plugging of ECCS suction strainers at the Perry Nuclear Power Plant and Barsebäck plant in Sweden. This is discussed in NRC Bulletin (BL) 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 11, 1994. The staff had originally proposed that the advanced designs should have the ability to backflush the suction strainers, which is similar to the resolution taken in Sweden for the Barsebäck plant. However, in evaluating the events, the staff decided to increase the sump sizing criteria, rather than requiring a backflush capability. As a result, in the "Advance Copy of Safety Evaluation Report for the Advanced Boiling Water Reactor (ABWR)," which was sent to General Electric in the staff letter dated December 30, 1993, the staff stated that an acceptable resolution for the advanced designs would be to size the ECCS suction strainers in accordance with RG 1.82, Revision 1, but with the factor of three screen area margin.

The staff had conducted a qualitative assessment of the risk associated with not applying the three-times multiplier for the ABWR design. This was applied to the System 80+ design. The assessment showed that the incremental risk is marginal unless very pessimistic assumptions are used; however, because of the uncertainties in the staff's knowledge of the severity of this phenomenon on the design-basis LOCA, the staff has decided to take a conservative position. For operating plants, the staff issued BL 93-02, Supplement 1, which requested interim compensatory measures to minimize the potential for the loss of ECCS suction pressure during a LOCA. Further analysis is required to assess the impact of non-fibrous debris on the potential for ECCS pump head loss because the staff has not bounded the magnitude of this issue.

Therefore, it is prudent to consider a more conservative position (i.e., the three-times screen area multiplier) to ensure compliance of the System 80+ design with 10 CFR 50.46. This position is in conformance with the Commission's advance reactor policy goal of providing a greater margin of safety for the next generation of reactor designs, such as the System 80+ design.

The staff has reassessed the potential impact of clogging of the ECCS suction strainers on advanced light water reactors (ALWRs). The staff concludes that the System 80+ meets the staff's position on Issue A-43 which requires that all ECCS suction strainers be sized to three times the area that would be calculated based on RG 1.82 (Rev. 1) for all LOCA scenarios.

In the DSER, the staff concluded that the proposed resolution of Issue A-43 for the System 80+ design was in conformance with SRP Section 6.2.2 (Rev. 4) and RG 1.82 (Rev. 1), and acceptable pending the following actions:

(1) resolution of the open and confirmatory items in DSER Section 5.4.3 concerning (a) the potential vortex formation as part of shutdown risk review and (b) capability of shutdown cooling pumps to continue pumping subject to possible air and other effects.

(2) an analysis, necessary design enhancements, or both, to conclude the capability of the CSS pumps to continue pumping when subjected to possible air, debris, or other effects such as particulate ingesting on pump seal and bearing systems.

The staff finds that the open and confirmatory items in Section 5.4.3 of this report, identified in Items (1)(a) and (1)(b) above, have been resolved. As discussed above and in Section 19.3.2.3 of this report, the two CSS pumps take suction from the SIS suction headers. Therefore, the above resolution is applicable to the CSS pumps and the shutdown cooling pumps, which are functionally interchangeable with the CSS pumps.

Therefore, based on the above, the staff finds ABB-CE's response to this issue is acceptable and Issue A-43 is resolved for the System 80+ design.

Issue A-44: Station Blackout

Issue A-44, in NUREG-0933, addressed the likelihood and duration of the loss of all ac power at the site (i.e., station blackout(SBO)), and the potential for severe core damage after the SBO. An SBO could be an important contributor to the total risk from an nuclear power plant. This issue was resolved in 1988 with the publication of 10 CFR 50.63 (53 FR 23203) and RG 1.155, Station Blackout."

In CESSAR-DC Section 20.2.67, ABB-CE lists the improved design features and electrical systems in System 80+ to ensure a safe shutdown of the reactor during an SBO. In CESSAR-DC Sections 8.1.4.2 and 8.3.1.1.5, ABB-CE describes the alternate ac (AAC) power source that is designed to power one safety-related load division and its corresponding essential non-safety-related load bus within 2 minutes of an SBO. The staff has evaluated SBO for the System 80+ design in Section 8.5 of this report.

On the basis of the staff's conclusions in Section 8.5 of this report, Issue A-44 is resolved for the System 80+ design.

Issue A-45: Shutdown Decay Heat Removal Requirements

Issue A-45, in NUREG-0933, addresses the safety adequacy of the decay heat removal (DHR) function in an operating LWR and assesses the value and impact of alternate measures to improve the overall reliability of the RHR function. In response to the DHR PRA study, the NRC established a goal that core damage due to failure of the DHR function should be less than 1 x 10^{-5}/reactor-year, as identified in NUREG-0933. This goal should be demonstrated by a Level 1 SCS PRA for the System 80+ design.

ABB-CE has conducted a Level 1 PRA for the System 80+ design, which includes an assessment of the core-damage frequency (CDF) failure of the SCS. The PRA determines the CDF attributable to internal initiating events such as SG tube rupture (SGTR) and station blackout (SBO), as well as external events, such as tornados and earthquakes. The PRA is in CESSAR-DC Chapter 19. The PRA showed that the CDF for failure of the SCS capability, along with failure of other systems included in the core-damage sequences, is lower than the staff requirements noted above. The staff reviewed the PRA findings of ABB-CE in assessment of the compliance of the System 80+ design with the NRC guidance regarding the performance goal for the SCS and found that the PRA submittals are acceptable (see Section 19.1 of this report for the evaluation). In addition, the staff concluded in Section 19.3 of this report that ABB-CE's shutdown risk assessment provides reasonable assurance that the System 80+ design will significantly reduce the shutdown risk and is acceptable.

Therefore, Issue A-45 is resolved for the System 80+ design.

In the DSER, the staff did not include its evaluation of ABB-CE's PRA in assessing the compliance of the System 80+ design in meeting the staff's performance goals for the SCS. The staff stated that it would address this compliance later in this report and designated this as DSER Open Item 20.1-6. Therefore, as discussed above, DSER Open Item 20.1-6 is resolved.

Issue A-47: Safety Implications of Control Systems

Issue A-47, in NUREG-0933, concerns the potential for accidents or transients becoming more severe as a result of control systems failures. Within this issue, the staff performed an in-depth review of non-safety-related control systems and assessed the effect of control system failures on plant safety.

Non-safety-grade control systems are not relied on to perform any safety functions, but they are used to control plant processes that could have a significant impact on plant dynamics. For the resolution of Issue A-47, the NRC evaluated the effects of control system failures on PWR reference plants, including a design subjected to single and multiple control system failures during automatic and manual modes of operation. The staff raised two concerns related to the design: (1) SG overfill and (2) reactor core heat removal to cold shutdown after a small-break LOCA (SBLOCA), without overcooling the reactor vessel. The NRC issued GL 89-19, "Request for Action Related to Resolution of USI A-47, Pursuant to 10 CFR 50.54(f)," dated September 20, 1989, which required all operating PWR plants and plants under construction to provide automatic protection from SG overfill by the main feedwater (MFW) system.

The first acceptance criterion for the resolution of Issue A-47 is that the plant shall have, as a minimum, control-grade protection against SG overfill by MFW, and by TSs and plant operating procedures to ensure in-service verification of the availability of the overfill protection, in accordance with GL 89-19.

In CESSAR-DC Section 20.2.69, ABB-CE states that the System 80+ design includes a MFW isolation system to protect the SGs from being overfilled. The system includes redundant, remotely operated isolation valves in each MFW line to each SG. The valve actuation system comprises redundant trains A and B, with physically and electrically separate I&Cs, so that a failure in one train will not impair the actions of the other train. The MFW isolation valves are automatically actuated by a main steam isolation signal (MSIS) from the ESFAS. High SG water level, in a 2-out-of-4 logic, is one of the initiators of the MSIS. The MFW isolation valves can be in-service tested in accordance with ASME Code, Section XI, Subsection IV. A TS establishes testing requirements for the valve actuation system. These requirements will also be incorporated into the plant maintenance procedures.

The second criterion is that the SI pressure capability should be greater than 8791 kPa (1275 psia) and EFW should be automatically initiated on a low SG water level signal.

In an SBLOCA, high-pressure SI in the System 80+ design is delivered at a pressure that exceeds the 8791 kPa (1275-psia) requirement. The System 80+ design also incorporates a safety-grade emergency feedwater system (EFWS) that is automatically actuated by an emergency feedwater actuation signal (EFAS) from the ESFAS, or by an AFAS from the APS. There is one EFAS for each SG, initiated by low SG water level in a 2-out-of-4 logic. The EFWS, in conjunction with safety-grade atmospheric steam dump valves, provides an independent means of RHR from the RCS via the secondary system until the RCS pressure and temperature permit actuation of the SCS. An ESFAS high SG water level interlock will isolate EFW to preclude SG overfill. The RCS depressurization rate is manually controlled by the operator from the control room to prevent overcooling of the reactor vessel, by throttling the EFW or using the pressurizer auxiliary sprays or both.

In addition to these features, ABB-CE provides emergency operations guidelines (EOGs), as discussed in Chapter 18 of this report, to the owner/operator for preparing emergency operating procedures (EOPs) detailing the actions to be taken by the plant operators in the event of an SBLOCA. See the resolution of Issue I.C.1 in Section 20.4 of this chapter.

ABB-CE has acceptably addressed Issue A-47 and GL 89-19 and, therefore, Issue A-47 is resolved for the System 80+ design.

Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Issue A-48, in NUREG-0933, was to consider additional hydrogen control and mitigation systems for power reactors with small containment structures. Although hydrogen control measures in connection with a design-basis LOCA (DBLOCA) had been required by 10 CFR 50.44 well before the Three Mile Island, Unit 2 (TMI-2) accident, metal-water reactions generated hydrogen during the accident in excess of the amounts specified in 10 CFR 50.44.

In response to the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34 and 50.44. 10 CFR 50.34(f) requires a hydrogen control system based on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment, or a postaccident atmosphere that will not support hydrogen combustion. Only those plants whose CPs had not been issued at the time of the TMI-2 accident are covered by this rule.

In CESSAR-DC Table 20.1-1, ABB-CE originally considered that this issue was applicable to the System 80+ design. In CESSAR-DC Amendment U, upon further review, ABB-CE concluded that Issue A-48 was not applicable because the issues had been superseded.

COL Action Item 20.2-7 in the DSER identified the requirement for the staff to review relevant plant-specific design features regarding combustible gas control for conformance to 10 CFR 50.34(f) when an application is received. The staff's review of this issue concludes that the System 80+ design meets the requirements of SECY-90-016 ("Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990) and 10 CFR 50.34(f) for hydrogen control (see Sections 19.2.3.3.1 and 19.2.3.6.1 of this report). Therefore, DSER COL Action Item 20.2-7 is resolved.

Based on this, Issue A-48 is resolved for the System 80+ design. As stated in NUREG-0933, this issue was integrated into the resolution of Issue 121. See also the discussion of Issue 121 in Section 20.3 of this chapter.

Issue A-49: Pressurized Thermal Shock

The neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of these materials. The staff's concern is the possibility of vessel failure due to a severe pressurized overcooling event, or thermal shock. This was designated Issue A-49 in NUREG-0933.

As noted in NUREG-0933, this issue was resolved and new requirements were established in 10 CFR 50.61 and incorporated into RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors." As discussed in Section 5.2.2.3 of this report, the reactor vessel beltline materials proposed by ABB-CE for the System 80+ design meet the requirements of 10 CFR 50.61. Compliance with this rule is an acceptable basis for the resolution of this issue and, therefore, Issue A-49 is resolved for the System 80+ design.

Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

In NUREG-0933, this issue was divided into the following two parts which were separately evaluated:

Part I - Ductility of Two-Way Slabs and Shells

Part I of Issue B-5 was defined in NUREG-0471, "Generic Task Problem Descriptions," dated June 1978, and addressed the lack of information related to the behavior of two-way reinforced-concrete slabs loaded dynamically in biaxial tension, flexure, and shear. The objective was to develop design requirements for concrete two-way slabs to resist loading caused by a LOCA or high-energy line break (HELB). An acceptable resolution to this issue is to apply the two-way reinforced-concrete slab analysis methods to adequately address dynamic loading in biaxial membrane tension, flexure, and shear due to a LOCA or HELB.

In CESSAR-DC Section 20.2.71, ABB-CE states that the methods in Appendix C of American Concrete Institute (ACI) 349-85, "Code Requirements for Nuclear Safety Related Structures," dated 1985, and Positions 10 and 11 of RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (other Than Reactor Vessel Containments)," (Rev. 1, October 1981), are used to treat the impactive and impulsive loads associated with a LOCA or HELB. Also, ABB-CE states that the containment piping analysis uses the LBB methodology to reduce the number of situations in which these loadings occur. The commitment to RG 1.142 was in response to staff RAI Q220.54 which is listed in Appendix B of this report and was also documented in ABB-CE's letter dated February 25, 1992.

The commitment by ABB-CE to the methods in Appendix C of ACI 349-85 and Positions 10 and 11 of RG 1.142 in the CESSAR-DC is acceptable. Based on this, Part I of Issue B-5 is resolved for the System 80+ design.

Part II - Buckling Behavior of Steel Containments

Part II of Issue B-5 was also identified in NUREG-0471 and addressed the lack of a well-defined approach for design evaluation of steel containment vessels subject to asymmetrical dynamic loadings that may be limited by the instability of the shell. An acceptable resolution to this issue is to address adequately the design loads, the asymmetrical vessel configurations associated with the presence of equipment hatches, and the factor of safety in determining allowable loadings.

In CESSAR-DC Section 20.2.71, ABB-CE states that this issue is resolved because the steel containment design satisfies the requirements in ASME Code, Section III and there is no asymmetric dynamic pressure from the layout and design of the reactor building. ABB-CE also states that the actual safety factor for the stability analysis is derived from a three-dimensional large deflection analysis taking into account imperfections and non-linear material properties.

In Section 3.8.2 of this report, the staff describes in detail its evaluation of the steel containment design and buckling load analysis. Based on the staff's conclusions in this section, Part II of Issue B-5 is resolved for the System 80+ design.

Issue B-17: Criteria for Safety-Related Operator Actions

Issue B-17, in NUREG-0933, involves the development of a time criterion for safety-related operator actions (SROAs), including a determination of whether automatic actuation is required. This issue also concerns PWR designs that require manual operations to accomplish the switchover from the injection mode to the recirculation mode following a LOCA.

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. Consequently, it becomes necessary to develop appropriate criteria for SROAs. The criteria would include a determination of actions that should be automated in lieu of operator actions and development of a time criterion for SROAs.

The review criteria for this issue are contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," and ANSI/ANS 52.2-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants." Plants should perform task analysis, simulator studies and analysis and evaluation of operational data to assess ESF and safety-related control system designs for conformance to the criteria. Where nonconformance is identified, modification of the design and hardware may be required.

In the revised operating experience report (OER) (CESSAR-DC Amendment Q), which is discussed in Sections 18.3 and 18.4 of this report, ABB-CE indicates that the requirement for automation of the switch from the injection mode to the recirculation mode is not applicable because the System 80+ design has an in-containment refueling water tank. ABB-CE notes that the System 80+ design has eliminated the switchover function. In CESSAR-DC Section 20.2.72, ABB-CE states that the System 80+ design does not require operator actions during the first 30 minutes for all DBEs. See Chapter 15 of this report. The staff finds the information provided by ABB-CE acceptable and, therefore, this issue is resolved for the System 80+ design.

In the DSER, the staff stated that this issue would be addressed in this report and designated the action incorrectly as DSER Action Item 20.1-19. The correct number was DSER Action Item 20.2-19. On the basis of this evaluation, DSER Action Item 20.2-19 is resolved.

Issue B-26: Structural Integrity of Containment Penetrations

Issue B-26, in NUREG-0933, addressed the adequacy of specific containment penetration designs regarding structural integrity, ISI requirements, and new surveillance or analysis methods applicable to containment penetrations that are identified as inaccessible. In 1984, after reevaluating this issue, the staff determined that the increase in occupational radiation exposure from additional inspections would negate the small potential risk reduction associated with the issue. As a result, the staff stated in NUREG-0933 that the issue was resolved and no new requirements were established.

In CESSAR-DC Table 20.1-1, ABB-CE categorizes this issue as not relevant to the System 80+ design because, based on the staff's evaluation discussed above, the issue was resolved with no new requirements established. ABB-CE's disposition of this issue is acceptable and, therefore, Issue B-26 is resolved for the System 80+ design.

Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems

Issue B-36, in NUREG-0933, was to revise the then-current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units. This issue was resolved by the issuance of Revision 2 of RG 1.52 for ESF ventilation filter units in March 1978, and Revision 1 of RG 1.140 for normal ventilation filter units, in October 1979.

In CESSAR-DC Section 20.2.73, ABB-CE states that the ventilation systems meet the RGs listed above. Sections 6-.4, 6.5.1, and 11.3 of this report discuss the compliance of the System 80+ design with the guidelines in RGs 1.52 and 1.140. This issue is resolved by the staff and there are no requirements to be resolved by ABB-CE.

The System 80+ design has the following ventilation systems which filter radioactivity under normal and postaccident conditions: control room, fuel building, nuclear annex and radwaste building, annulus ventilation, subsphere building, and containment cooling and ventilation. However, ABB-CE does not always take credit for these filtration units during accidents. The design of the atmosphere cleanup part of these systems is in accordance with either Revision 2 of RG 1.52 or Revision 1 of RG 1.140. ABB-CE commits to both these RGs in CESSAR-DC Table 1.8-1 and Section 20.2.73.

Based on the above, Issue B-36 is resolved for the System 80+ design.

Issue B-53: Load Break Switch

Issue B-53, in NUREG-0933, relates to the reliability of a load break switch or a circuit breaker that, in some plant designs, is relied on to isolate the plant's main generator from the grid following a turbine trip. This is to allow power to be fed in the reverse direction from the grid through the main transformer to Class 1E buses. This circuit is usually used as an immediate offsite power source for the Class 1E loads. A generator circuit breaker, when used, has the added requirement to ensure isolation of the generator during fault conditions for the purpose of providing the immediate offsite power source to the Class 1E buses as is required by GDC 17. In NUREG-0933, the staff stated that this issue was addressed with the issuance of Appendix A to SRP Section 8.2 in July 1983.

As discussed in Section 8.2.2 of this report, the immediate offsite power source for the unit auxiliary loads and Class 1E loads is provided by a backfeed through the main stepup transformer to the unit auxiliary transformer by disconnecting the main generator from the transmission network by a generator breaker. ABB-CE states in CESSAR-DC Section 20.2.74 that the generator circuit breaker used in the System 80+ design would be qualified in accordance with the guidance provided in Appendix A to SRP Section 8.2; therefore, this issue is resolved for the System 80+ design.

Issue B-56: Diesel Reliability

Issue B-56, in NUREG-0933, addressed EDG reliability. This safety issue was promulgated by a review of LERs that indicated that EDGs at operating plants were demonstrating an average starting reliability of approximately 0.94 per demand.

The reliability of EDGs is one of the main factors affecting the risk of core damage from a SBO event. Thus, attaining and maintaining high reliability of EDGs at nuclear power plants is a major contributor to the reduction of the probability of SBO. In RG 1.155, "Station Blackout," the staff recommends an EDG reliability program that has the capability to achieve and maintain the EDG reliability levels in the range of 0.95 per demand or better to cope with SBO.

This issue was resolved by the issuance of RG 1.160, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and RG 1.9, Revision 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Onsite Electrical Power Systems at Nuclear Power Plants." RG 1.160 endorses NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

In CESSAR-DC Section 20.2.75, ABB-CE discusses the application of Issue B-56 to the System 80+ design. ABB-CE commits to conform to the guidance of RG 1.9, Revision 3, and states that the EDGs will have a maintenance program in accordance with the Maintenance Rule (10 CFR 50.65) and guidance in RG 1.160 to monitor diesel generator performance. This satisfies the requirements of this issue, and ABB-CE's resolution of Issue B-56 for the System 80+ design is acceptable.

The staff stated in the DSER that ABB-CE's proposed technical resolution of this issue would be evaluated in this report (DSER Open Item 20.2-20).
Based on the above review, DSER Open Item 20.2-20 is resolved.

Issue B-60: Loose-Parts Monitoring Systems

Applicants for CPs and operating licenses are required to commit to establishing a loose-parts detection program. The program is established to detect loose metallic parts in the RCS at an early stage. Early detection can give the time required to avoid or mitigate damage to, or malfunction of, safety-related primary system components. The NRC had developed hardware and operational criteria for loose-parts detection systems. These criteria are in Revision 1 of RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," which NRC issued in May 1981.

Issue B-60, in NUREG-0933, was to resolve any outstanding issues related to the implementation of Revision 1 of RG 1.133, including the development of staff positions and guidance with respect to upgrading loose-parts detection systems at operating facilities. This issue was resolved without any new requirements and, therefore, the guidelines for an acceptable loose-parts monitoring system (LPMS) in Revision 1 of RG 1.133 satisfactorily resolve this issue.

The resolution of Issue B-60 was incorporated into Item II.7 of SRP Section 4.4, which requires that the design description and proposed procedures for use of the LPMS be consistent with the guidance in RG 1.133.

The System 80+ design includes an LPMS to detect the presence of loose parts in the RCS. The LPMS is described in CESSAR-DC Sections 7.1.2.30 and 7.7.1.6.3.

LPMS sensors are installed at the locations given in CESSAR-DC Table 7.7-4. These locations correspond to natural collection regions for loose parts in the primary system and the secondary side of the SG. Two sensors are at each natural collection region and their associated cabling and amplifiers are physically separated. Signals from the sensors are routed via high-temperature, low-noise cable to in-containment charge amplifiers. The charge amplifier output is transmitted to alarm units within the equipment room. The alarm unit compares the peak value of the accelerometer output to a predetermined threshold and provides an alarm to the control room annunciator and plant computer systems. The LPMS is designed to be consistent with the guidance in RG 1.133 (Rev. 1). The staff has reviewed the LPMS design for the System 80+ plant by comparing it with the systems used at other plants, taking into account pertinent differences. As stated in Section 4.4.3 of this report, the staff concludes that an acceptable LPMS will be implemented for the System 80+ plants. Therefore, ABB-CE's resolution of this issue for the System 80+ design is acceptable.

In DSER Confirmatory Item 20.2-2, the staff stated that limiting conditions for operation (LCOs) and surveillance requirements for the LPMS would be included in the CESSAR-DC TSs in accordance with RG 1.133. However, to be consistent with the improved ABB-CE Standard TSs, the LPMS is not included in the TSs for System 80+, as discussed more fully in Section 4.4.3 of this report. Therefore, Confirmatory Item 20.2-2 is resolved.

Issue B-61: Allowable ECCS Equipment Outage Periods

Issue B-61, in NUREG-0933, was raised to establish surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for the TSs. The present TS-allowable equipment outage intervals and test intervals were determined primarily on the basis of engineering judgment. Studies performed by the NRC on operating reactors indicated that from 30 to 80 percent of the ECCS unavailability was due to testing, maintenance, and allowed outage periods. Therefore, by optimizing the allowed outage period and the test and maintenance interval, the equipment unavailability and public risk can be reduced.

Because the NRC has not completed its evaluation of this issue, the initial LCOs for a future plant design may continue to be based on current industry practice without prejudicing later optimization when the methods and requirements have been confirmed. The LCO surveillance periods and outage times shall be accounted for in the overall plant PRA as required by 10 CFR Part 52. Any subsequent proposed changes to the provisions in the LCOs for ECCS surveillance shall be demonstrated to be within the results of an existing PRA.

In CESSAR-DC Section 20.2.77, ABB-CE states that the System 80+ design evolved from the System 80 design with the incorporation of design enhancements to improve the operation and safety of the plant, and the most significant advances are in the area of ESFs. These include a four-train system for high-pressure SI drawing water from an IRWST, which permits long-term recirculation without a changeover of water sources (also true for the CSS), a dedicated safety-grade EFWS, and an integrated containment spray and SCS. Also included are a safety depressurization system (SDS), and an AAC power source to help cope with loss-of-power events.

The LCOs for the System 80+ design (see CESSAR-DC Chapter 16) are developed from the System 80 LCOs in the TSs for the Palo Verde Nuclear Generating Station, taking into account the differences in design and safety improvements alluded to above. The System 80 LCOs were in turn developed from the experience with similar systems and components during many years of operation at previous CE nuclear power plants. Thus, the System 80+ design LCOs for the surveillance and outage times for safety equipment are consistent with the same general body of component availability experience that is used as input to the System 80+ design PRA. (See CESSAR-DC Chapter 19.)

The PRA uses a system fault tree approach to quantify system accident sequences that result in severe core damage. Data related to ESFs that are used in the quantification include

- component failure rates
- component repair times and maintenance frequencies
- component inspection and test times and frequencies
- allowable equipment outage times

The data are used in accordance with NUREG/CR-2815 (U.S. NRC, "Probabilistic Safety Analysis Procedures Guide," January 1984). The basic failure rate data are obtained from the EPRI ALWR Utility Requirements Document ("Advanced Light Water Reactor Requirements Document - Chapter 1: Overall Requirements, Appendix A: PRA Key Assumptions and Groundrules," Draft, April 1987) supplemented with data from the National Reliability Evaluation Program (NREP) Generic Data Base (E.G. and G. "Generic Data base for Data and Models Chapter of the National Reliability Evaluation Program (NREP) Guide," EGG-EA-5887, June 1982) and other nuclear sources. Maintenance and repair times are calculated as outlined in NUREG/CR-2815. The inspection and test times and frequencies are as specified in the System 80+ LCOs (see CESSAR-DC Chapter 16).

The PRA demonstrates that the System 80+ design meets the industry goal of 1.0 x 10^{-5} CDF/reactor-year for future reactors and indicates that the initial LCOs are consistent with this goal. The COL applicant may refine the LCOs to further reduce risk or increase operational flexibility provided that the resulting overall risk is shown to be within the above goal.

The staff's evaluation of the System 80+ TSs is in Section 16 of this report. Based on this evaluation and ABB-CE's response to this issue, Issue B-61 is resolved for the System 80+ design.

Issue B-63: Isolation of Low-Pressure Systems Connected to the RCPB

Several systems connected to the RCPB have design pressures that are considerably below the RCS operating pressure. The NRC has required that valves forming the interface between these high- and low-pressure systems have sufficient redundancy to ensure that the low-pressure systems are not subjected to pressures beyond their design limits.

Recently, there has been discussion about the adequacy of the isolation of low-pressure systems that are connected to the RCPB. Earlier reviewers have concentrated on ensuring isolation of the RHR system, which is a low-pressure system in almost all PWRs and BWRs. Current reviews of license applications for new plants are based on guidelines in the SRP supplemented by the staff position, "Leak Testing of Pressure Isolation Valves," in SRP Section 3.9.6 (Rev. 2).

Issue B-63, in NUREG-0933, was to assess the isolation capabilities of low-pressure systems by reviewing a representative operating plant for each nuclear steam supply system (NSSS) vendor, including ABB-CE. Each low-pressure system connected to the RCPB and penetrating the containment would be examined. In April 1981, an order was issued to licensees for all operating reactors to comply with the requirements of the resolution for Event V configurations. All other configurations were addressed by SERs on inservice testing and were issued as license amendments.

In CESSAR-DC Section 20.2.78, ABB-CE state that because of the importance of the interface between high-pressure (HP) and low-pressure (LP) safety-related systems, all pressure-containing components used in the System 80+ design identified as Safety-Class 1, 2, or 3 (including all HP-to-LP safety-related system boundary valves, such as SCS isolation valves) are designed, manufactured, and tested in accordance with ASME Code, Section III (see CESSAR-DC Section 3.2.2). CESSAR-DC Table 3.2-2 provides a cross-reference between safety class and code class. Furthermore, ABB-CE provides in CESSAR-DC Section 3.9.6.2.4 a list of RCS pressure isolation valves (PIVs). ABB-CE states that those PIVs will be leak-rate tested in accordance with CESSAR-DC Table 3.9-15 and the surveillance requirements specified in the TS 3.4.13.1. The staff's evaluation of the RCS PIVs leak testing requirements for the System 80+ has been found to be acceptable as discussed in Section 3.9.6.2.4 of this report.

Based on the above, Issue B-63 is resolved for the System 80+ design.

Issue B-66: Control Room Infiltration Measurements

The control room area ventilation systems and control building layout and structures are reviewed to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases and that the control room can be maintained as the backup center from which technical personnel can safely operate during an accident. A key parameter affecting control room habitability is the rate of air infiltration into the control room. Current estimates of these rates are based on data relating to buildings that are substantially different from typical control room buildings in nuclear power plants.

Issue B-66, in NUREG-0933, was to facilitate compliance with the following staff requirements and guidance on control room habitability: (1) GDC 19 and (2) SRP Sections 6.4, "Control Room Habitability Systems," and 9.4.1, "Control Building Ventilation Systems." Additional experimentally measured air exchange rates of operating reactor control rooms resulted in Revision 2 of SRP Section 6.4. See also the resolution of Issues 83 and III.D.3.4 for the System 80+ design in Sections 20.3 and 20.4, respectively, of this chapter.

In CESSAR-DC Section 20.2.79, ABB-CE states that the System 80+ control room ventilation system design provides continuous pressurization of the room to prevent entry of dust, dirt, smoke, and radioactivity originating from outside the room. Filtered outdoor air for pressurization is taken from either of two locations so that a source of uncontaminated air is available. Each intake location is monitored for radioactivity, toxic gases, and products of combustion (see CESSAR-DC Section 9.4.1). In the event of an outside air contamination signal from the control room intake radiation monitors, or upon receiving a safety injection actuation signal (SIAS), component control logic will automatically close the more contaminated intake and divert the control room intake and recirculation flows via the designated control room filtration unit. Identification of potential hazardous gas sources and releases at or in the vicinity of a specific plant site, and analysis of the resulting concentration in the control room, are the responsibility of the COL Applicant.

Outside air will flow through the filter train to the control room to maintain a positive pressure with respect to the surrounding area. Room air temperatures are maintained at habitable levels by internal recirculation cooling. The habitability systems are able to reliably perform their functions during emergency conditions due to the design features of the systems (see CESSAR-DC Sections 6.4 and 9.4.1).

The System 80+ control room infiltration measurements were evaluated by the staff and found acceptable in accordance with SRP Sections 6.4 and 9.4.1, in Sections 6.4 and 9.4.1 of this report. Therefore, Issue B-66 is resolved for the System 80+ design.

Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Issue C-1, in NUREG-0933, was developed because of concerns regarding the long-term capability of hermetically sealed instruments and equipment that must function in postaccident environments. Certain classes of instrumentation incorporate these seals. When safety-related components within the containment must function during post-LOCA conditions, their operability is sensitive to the ingress of steam or water.

ABB-CE addressed this issue in CENPD-255-A (Rev. 3), CESSAR-DC Section 20.2.57 for Issue A-24, and CESSAR-DC Section 3.11.

Resolution of open and confirmatory issues in the DSER for CESSAR-DC Section 3.11 and staff concerns with

Issue A-24 in the DSER were required to resolve all concerns with Issue C-1. The open and confirmatory issues in the DSER for CESSAR-DC Section 3.11, and the NRC staff's concerns with Issue A-24 have been resolved as discussed above in Issue A-24 in this section. The NRC staff previously reviewed and approved CENPD-255-A as discussed in Section 3.11 of this report.

Therefore, Issue C-1 is resolved for the System 80+ design.

Issue C-2: Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure

Issue C-2, in NUREG-0933, was to develop a code to be used for the analysis of containment pressure response with and without the effects of vacuum breakers or control systems for the inadvertent spray event. Inadvertent operation of containment sprays can result in a rapid depressurization of the containment building. Where containment external design pressure may be exceeded, plants have been provided with vacuum breakers or control system interlocks to prevent this condition. The resolution of this issue was to require licensees to perform analyses of containment depressurization due to inadvertent spray operation, and the staff would review these analyses in accordance with SRP Section 6.2.1.1.

The staff reviewed ABB-CE's analyses of containment depressurization caused by inadvertent spray operation in accordance with SRP Section 6.2.1.1. A detailed discussion of the staff's review is in Section 6.2.1.1 of this report.

Therefore, based on the staff's conclusions in Section 6.2.1.1 of this report, Issue C-2 is resolved for the System 80+ design.

Issue C-4: Statistical Methods for ECCS Analysis

Issue C-4, in NUREG-0933, addresses the statistical methods used for performance evaluation of the ECCS during a LOCA. In accordance with the requirements of 10 CFR 50.46 as amended on September 16, 1988, the NRC requires that the LOCA analyses for license applications use either the 10 CFR Part 50 (Appendix K) evaluation models or the statistical (realistic) models, including the uncertainty of calculation in the adverse direction. The realistic models must be supported by applicable experimental data. Uncertainties in the realistic models and input must be identified and assessed so that uncertainty in the calculated results can be estimated.

ABB-CE used the approved 10 CFR Part 50 (Appendix K) evaluation models to perform the LOCA analysis for the System 80+ ECCS design. The LOCA analysis is discussed in CESSAR-DC Section 6.3.3. The staff reviewed the LOCA analysis for the System 80+ design and concludes in Section 15.3.7 of this report that the LOCA analysis is acceptable because the approved Appendix K evaluation models were used for analysis, and the analytical results complies with the ECCS performance acceptance criteria specified in 10 CFR 50.46.

Therefore, based on the staff's conclusions in Section 15.3.7 of this report, Issue C-4 is resolved for the System 80+ design.

Issue C-5: Decay Heat Update

Issue C-5, in NUREG-0933, addressed the specific decay heat models for the LOCA analysis models. In accordance with the requirements of 10 CFR 50.46 as amended on September 16, 1988, the LOCA analyses for license applications should use either the 10 CFR Part 50 (Appendix K) models, or the realistic models supported by applicable experimental data and including uncertainty of calculation in the adverse direction. When Appendix K models are used, the decay heat generation function should be based on ANS 5.0, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," plus a 20-percent uncertainty factor. When realistic models are used, the decay heat function in ANS 5.1, "Decay Heat Power in Light Water Reactors," is acceptable for licensing applications.

In CESSAR-DC Section 6.3.3, ABB-CE analyzes the LOCA for the System 80+ design. The staff reviewed the LOCA analysis and prepared the evaluation in Section 15.3.7 of this report. The 10 CFR Part 50 (Appendix K) models were used for the LOCA analysis. The required decay heat function in ANS 5.0 with inclusion of 20-percent uncertainty factor was used. Therefore, the use of the decay heat function complies with the requirements of 10 CFR 50.46 and is acceptable.

Therefore, Issue C-5 is resolved for the System 80+ design.

Issue C-10: Effective Operation of Containment Sprays in an LOCA

Issue C-10, in NUREG-0933, addressed the effectiveness of various containment sprays to remove airborne radioactive material that could be present within the containment following a LOCA. This was expanded to include the possible damage to equipment located within the containment due to an inadvertent actuation of the sprays. This issue was resolved by SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," which references ANSI/ANS 56.5-1979, "PWR and BWR Containment Spray System Design Criteria."

The staff evaluated the CSS against the requirements in SRP Section 6.5.2, in Section 6.5 of this report. Based on the staff's conclusions in this section, Issue C-10 is resolved for the System 80+ design.

Issue C-12: Primary System Vibration Assessment

Issue C-12, in NUREG-0933, addressed the potential for detrimental effects on the primary system from flow-induced vibrations, vibrations that occur from operation of the primary system pumps, or from other causes. Of concern is the possibility that excessive vibration could lead to premature failure of RCS components, causing damage to the internal components of the reactor vessel and, potentially, interference with the operation of the control rod system. The staff has concluded that SRP Section 3.9.2 ("Dynamic Testing and Analysis of Systems, Components, and Equipment") and RGs 1.20 ("Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing") and 1.133 ("Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors") provide sufficient basis for resolution of this issue.

In CESSAR-DC Section 20.2.85, ABB-CE states that guidance from SRP Section 3.9.2 and RG 1.133 is incorporated into the system and component design process. The SRP requirements include acceptance of a vendor's prototype plant results along with the startup program which satisfies RG 1.20 (Rev. 2). ABB-CE addresses this issue by considering vibration during the design phase and then by monitoring the vibration during plant startup and operation. The System 80 plants at Palo Verde are considered the prototype plants for the System 80+ design and, as such, experience from the startup of these plants is included in the System 80+ design. The System 80+ design also includes a vibration and leak monitoring system called the NSSS integrity monitoring system (NIMS), which consists of the following three subsystems: (1) internals vibration monitoring system (IVMS), (2) acoustic leak monitoring system (ALMS), and (3) LPMS. The IVMS and LPMS provide data from which changes in motion of the reactor vessel's internal components can be detected, as well as the presence of a loose part within the RCS pressure boundary.

ABB-CE's proposed approach in addressing this issue conforms with SRP Section 3.9.2, and RGs 1.20 (Rev. 2) and 1.133. Therefore, the proposed approach is acceptable and Issue C-12 is resolved for the System 80+ design.

Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Issue C-17, in NUREG-0933, was to develop criteria for acceptability of radwaste solidification agents to properly implement a process control program for packaging diverse radioactive plant wastes for shallow land burial. There are no current criteria for a finding of acceptability of solidification agents.

As stated in NUREG-0933, the Commission issued 10 CFR Part 61 on licensing requirements for land disposal of radioactive waste, including Section 61.56 which addresses acceptable waste characteristics. Also, BTP ETSB 11-3 was developed by the staff to be part of SRP Section 11.4, "Solid Waste Management Systems," and provide design guidance for solid waste management systems (SWMSs) to be used at LWRs. Therefore, this issue has been resolved for implementation at nuclear power plants.

The SWMS for the System 80+ design is evaluated in Section 11.4 of this report, where it is stated that the Part 61 requirements are a COL Action item (See CESSAR-DC Section 11.4.1.1, Item F) and the COL applicant is responsible for developing operating procedures to processing the wet solid radwastes to ensure that the Part 61 requirements are met. It is recognized that the development of a SWMS process control program and procedures is the responsibility of the COL applicant.

The staff concludes in Section 11.4 of this report that the SWMS can comply with 10 CFR Part 61, but the demonstration of such compliance is within the scope of the COL applicant. ABB-CE also concludes that is an operational issue for the COL applicant in CESSAR-DC Table 20.1-1. This is included in COL Action Item 11.4-1.

Based on the above, Issue C-17 is resolved for the System 80+ design.

20.3 New Generic Issues

Except for Issue 84 in this section, the new generic issues of NUREG-0933, were evaluated against the System 80+ design in this section:

- for the design to comply with 10 CFR 52.47(a)(1)(iv) and 10 CFR 50.34(f) in terms of ABB-CE addressing USIs, GSIs, and EMI Action Plan Items
- because ABB-CE states in CESSAR-DC Table 20.1-1 that the issue applied to the design

The staff also decided to include a discussion of Issue 84 for the System 80+ design.

Issue 3: Setpoint Drift in Instrumentation

Issue 3, in NUREG-0933, addressed the drift in the set points of instrumentation beyond the limits in the TS. Safety-related I&C systems use setpoints as a means of determining when to initiate a safety function. If the setpoint drifts, the actual value of the measured parameter at which a particular action is specified to occur will be altered. This can delay the initiation of a safety function.

The staff addressed this concern in RG 1.105 (Rev. 2), "Instrument Setpoints for Safety-Related Systems," which endorses the Instrument Society of America (ISA) standard ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants."

The acceptance criteria for resolving this issue are that safety-related I&C systems that use setpoints as a means of initiating their safety functions shall (1) establish and maintain the setpoints using the guidance in RG 1.105, (Rev. 2) (with the exception of ISA S67.04-1982), and (2) conform to the criteria in ISA S67.04-1987. Specifically, a setpoint shall be established so that its selection shall allow sufficient margin between the setpoint and the TS limit to account for the expected environmental conditions and other appropriate inaccuracies.

In CESSAR-DC Section 20.2.1, ABB-CE states that setpoint drift will be detected by periodic surveillance, through automatic testing of the plant protection system (PPS) bistable trip functions, and by monitoring the PPS setpoints by the data processing system (DPS). This is discussed in CESSAR-DC Sections 7.2.1.19 and 7.7.1.8.2.I. ABB-CE also states that the setpoints used to initiate plant safety functions are established and maintained following the requirements in ISA-S67.04-1987 and meet the guidance in RG 1.105 (Rev. 2).

The staff notes that the bistable functions are implemented in software; consequently, setpoint drift will not occur in the bistable functions. Instrument drift is detected through the on-line continuous signal validation with periodic calibration.

Since ABB-CE has committed to conform to ASI-S67.04-1987 and RG 1.105, Revision 2, Issue 3 is resolved for the System 80+ design.

Issue 14: PWR Pipe Cracks

Issue 14, in NUREG-0933, addressed cracking in PWR non-primary (i.e., secondary) piping systems as a result of stress corrosion, vibratory and thermal fatigue, and dynamic loading. Cracking in PWR non-primary system piping could lead to a decrease of the system functional capability and could possibly result in such situations as degraded core cooling. This issue deals with occurrences of MFW line cracking in certain Westinghouse and Combustion Engineering PWRs. In September 1980, the PWR Pipe Study Group completed its investigation of the issue and published its findings in NUREG-0691, ("Investigation and Evaluation of Cracking Incidents in Piping of Pressurized Water Reactors," dated September 1980). This report provides conclusions regarding systems safety and recommends technical solutions to the issue. The staff considered augmented inspections and inspection requirements, and concluded that they had low risk-reduction value. Therefore, this issue was resolved and no new requirements were established.

ABB-CE states in CESSAR-DC Section 20.2.2 that the high-energy secondary piping systems will be designed, manufactured, constructed, tested, and inspected in accordance with accepted industry codes and standards, and will meet the intent of the relevant guidance in SRP Chapters 3 and 10, and RG 1.26 (Rev. 3), "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

In the DSER, the staff requested that ABB-CE also reference SRP Section 6.6, "Inservice Inspection of Class 2 and 3 Components," in CESSAR-DC Section 6.6 for all ASME Class 2 and Class 3 piping systems. This was designated DSER Open Item 20.2-1. ABB-CE in Amendment L referenced SRP Section 6.6, in CESSAR-DC Section 6.6. Therefore, DSER Open Item 20.2-1 is resolved and the NRC staff concluded that this design approach is adequate in ensuring the structural integrity of non-primary piping. This is discussed in CESSAR-DC Appendix 3.9A and Sections 8.12 and 6.6 of this report.

Based on the above, Issue 14 is resolved for the System 80+ design.

Issue 15: Radiation Effects on Reactor Vessel Supports

Issue 15, in NUREG-0933, addresses the potential for radiation embrittlement of reactor vessel support structures. Neutron irradiation of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil-ductility transition temperature (NDTT). As long as the operating environment in which the materials are used has a higher temperature than the material's NDTT, failure by brittle fracture is not expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT, that is, they become more susceptible to brittle fracture at the operating temperatures of interest. This effect has to be accounted for in the design and fabrication of reactor vessel support structures.

As stated in NUREG-0933, this issue has a high priority ranking and requirements have not been issued by NRC.

ABB-CE discusses the resolution of this issue for the System 80+ design in CESSAR-DC Section 20.2.4. The reactor vessel support system consists of four vertical columns, which are located under the reactor vessel inlet nozzles. ABB-CE indicates that the design of the reactor vessel supports addresses irradiation effects (including low temperature and low neutron flux) and material embrittlement.

ABB-CE states that the reactor vessel supports are designed to accept normal, seismic, and branch-line pipe break loads. Irradiation effects are addressed in the fracture mechanics analysis of the columns which is performed using the philosophy of ASME Code, Section III, Appendix G to ensure that structural integrity is maintained. This fracture mechanics analysis addresses potential embrittlement and accident loads, including the SSE and large-break LOCA. ABB-CE contends that this analysis demonstrates that the structural integrity of the columns would be maintained, even if large cracks existed in the columns and they were subjected to the lowest possible temperatures and the maximum normal and SSE loadings. ABB-CE states that the sensitivity to uncertainty in the extent of the embrittlement is also addressed and the conservatism of this analysis is enhanced by the adoption of the LBB method in the System 80+ design basis.

Open items for this issue were identified in the discussion on this issue in the DSER. The resolution of each open item is listed below.

(1) ABB-CE has described in its letter dated December 23, 1992, the materials selected for the construction of the reactor vessel supports, limits on residual elements to minimize susceptibility to irradiation, limits on initial reference temperature and upper-shelf impact energy, and inspection requirements of supports during fabrication as follows:

The reactor vessel support columns are made of high-quality SA 508 steel, with additional restrictions on both its chemical composition and its postfabrication inspection. The specific chemistry restrictions are (a) maximum phosphorus, 0.012 percent per heat and 0.018 percent per product analysis and (b) maximum copper, 0.15 percent per heat and per-product analysis. Other chemical composition requirements consistent with SA 508 chemistry continue to apply.

The initial reference temperature (RT_{NDT}) for the unirradiated material is specified as 5 °C (40 °F), maximum. In actual practice, initial RT_{NDT} values of -12 to -1 °C (+10 to +30 °F) are typically achieved. The upper-shelf impact energy is specified to meet the fracture toughness requirements of ASME Code, Section III, Subsection NB-2300 at 5 °C (40 °F).

Postfabrication inspection is performed in accordance with ASME Code, Section III, Subsection NF, and ASME Code, Section II, Specification SA 508. Magnetic particle inspections in accordance with Method SA 275 are performed after final machining; forgings are ultrasonically inspected in accordance with Recommended Practice SA 388. These requirements should provide assurance of satisfactory material performance for a 60-year life. Based on this information, DSER Open Item 20.2-2 is resolved.

(2) ABB-CE has provided an estimated 60-year neutron fluence level at the reactor vessel support, which has been expressed in "displacement per atom (dpa)" to account for the spectrum of the neutron energy as discussed in NUREG/CR-5320, "Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants," January 1989, as follows:

The 60-year neutron fluence level is estimated to be 3.0 x 10^{18} neutrons/cm^2 (E > 1.0 Mev). This is based on an 80-percent capacity factor, that is, after 48 effective full-power years (EFPY). This fluence pertains to the surface of the support column facing the reactor, at core midplane. The actual fluence depends significantly on fuel management procedures employed over the life of the plant. The 60-year fluence is based on conservative physics calculations, and could exceed the fluence realized in actual practice by 30 percent or more. This fluence corresponds to approximately 0.0045 dpa. RT_{NDT} shifts for the reactor vessel supports can be reliably estimated using the methodology of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," while including conservative correction factors to account for the effects of temperature and active radiation parameters.

This should offer adequate assurance of satisfactory material performance for a 60-year life. Based on the information provided, DSER Open Item 20.2-3 is resolved.

(3) ABB-CE has described its procedures in estimating the extent of irradiation embrittlement as follows:

The effective fast fluence is used to calculate the irradiation-induced RT_{NDT} shift according to RG 1.99, Revision 2. RT_{NDT} shifts are calculated based on fluences at the locations of hypothetical crack tips within a structure. Crack tip fluences are somewhat lower than corresponding surface fluence values; the function describing the attenuation of fluence with depth is in RG 1.99, Revision 2, Section C, Part 1.1, Equation 3. For reactor vessel column support analyses, predictions of RT_{NDT} shift are based on SA 508 chemistry for which additional impurity restrictions have also been specified. The operating temperature range of the reactor vessel column supports at core midplane, well below 204 °C (400 °F), is then addressed. In RG 1.99, Revision 2, the staff states that temperatures below 274 °C (525 °F) should be considered to produce greater embrittlement than that predicted by its methodology and a correction factor should be used that is justified by reference to actual data. The available data indicate that the RT_{NDT} shift at temperatures below 204 °C (400 °F) is constant. The experimentally observed RT_{NDT} shifts below 400 °F (204 °C) exceed those at 288 °C (550 °F) by a factor of slightly more than 2. Accordingly, a conservative temperature correction factor of 2.25 is applied to the RT_{NDT} shift, as predicted by RG 1.99, Revision 2.

The reactor vessel column supports represent a redundant structure whose primary loading produces compressive stresses. For the reactor vessel column support analysis, the surface value for the initially specified maximum RT_{NDT} value of 5 °C (40 °F) is predicted to shift 90 °C (171 °F). This prediction was derived using material with the worst possible specified chemical composition exposed over 60 years (48 EFPY) to the 60-year fluence specified above. This prediction includes the conservative factor of 2.25 for colder temperatures. Considering that an actual material's chemistry and mechanical properties would stand up better, this analysis, with the redundant conservatism, provides reasonable assurance that the vessel supports will perform adequately for their 60-year life. Based on the information provided, DSER Open Item 20.2-4 is resolved.

(4) ABB-CE has submitted additional information on its fracture mechanics analysis, including the following assumptions and acceptance criteria:

The fracture mechanics evaluation of the reactor vessel column supports considers hypothetical cracks located at the core midplane, one on the side facing the reactor, and one on the side facing away. The method of ASME Code, Section XI is used to determine an applied stress intensity factor (K_i) associated with a hypothetical crack tip, using design condition static forces and moments in the column at core midplane, plus dynamic loadings from a SSE. Since the reactor vessel columns are fabricated from SA 508, Appendix G of ASME Code, Section III, is then invoked. Figure G-2210-1 in ASME Code, Section III, Appendix G, determines the minimum acceptable column temperature relative to an irradiated RT_{NDT}. The use of Appendix G has a further conservatism because the applied K_i associated with any primary membrane or primary bending stress is doubled before using Figure G-2210-1. Figure G-2210-1 then determines the minimum acceptable algebraic difference between the actual reactor vessel column temperature (RVCT), and the end-of-life (EOL) Irradiated RT_{NDT}. This algebraic temperature difference is then added to an additional margin requirement from 10 CFR Part 50, Appendix G. This is summarized in an equation as follows:

Initial RT_{NDT} + RT_{NDT} shift = Irradiated RT_{NDT} (I_{rtndt})

(where the RT_{NDT} shift is conservatively predicted with the factor of 2.25 for colder temperatures);

RVCT - I_{rtndt} >= Figure G-2210-1 + margin (10 CFR Part 50, Appendix G)

Since the ASME Code, Section III, Appendix G, requirement ultimately depends upon the dimensions of any hypothetical crack, the inequality above is then tested against crack dimensions that are increasingly larger until the inequality can no longer be satisfied; this determines a limiting crack size. This limiting crack size must be shown to be larger than the post-fabrication inspection flaw detection limits. This is an acceptable result for the reactor vessel column supports because the undetected flaws do not need to be repaired since there are smaller than the critical size. Any detected flaw must be repaired prior to certifying the reactor vessel column supports as acceptable. Based on this information, DSER Open Item 20.2-5 is resolved.

(5) The staff evaluated the application of LBB for the System 80+ design in Section 3.6.3 of this report. With LBB satisfied for selected piping systems, the dynamic effects from postulated pipe breaks in these piping systems are eliminated, therefore, DSER Open Item 20.2-6 is resolved.

The staff has determined that the proposal of ABB-CE is adequate in ensuring the structural integrity of the reactor vessel support system and, thus, is acceptable in resolving Issue 15 for the System 80+ design.

The COL applicant should verify that, on the basis of actual plant-specific values, its application meets the CESSAR-DC assumptions of reactor vessel support materials properties for the 60-year neutron fluence. This is included in COL Action Item 5.3.1-1 of Section 5.3.1 of this report.

Based on the above, Issue 15 is resolved for the System 80+ design.

Issue 22: Inadvertent Boron Dilution Events

Issue 22, in NUREG-0933, addressed the possibility of core criticality during cold shutdown conditions from an inadvertent boron dilution event. The acceptance criterion is that plants shall minimize the consequences of inadvertent boron dilution events by meeting the intent of SRP Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)." Specifically, the plant shall respond in such a way that the criteria regarding fuel damage and system pressure are met and the dilution transient is terminated before the shutdown margin is eliminated. If operator action is required to terminate the transient, redundant alarms must be in place and the following minimum time intervals must be available between the time when an alarm announces an unplanned dilution and the time shutdown margin is lost:

- during refueling (Mode 6) — 30 minutes
- during all other operating modes — 15 minutes

As stated in CESSAR-DC Section 20.2.4, ABB-CE performed a safety analysis for the System 80+ design which demonstrated that the consequences of an inadvertent boron dilution during cold shutdown are minimized. This analysis considered SRP Section 15.4.6 criteria, including design limits, a single failure in conjunction with moderate frequency events, and the impact of a single failure or operator error on fuel integrity and radiological dose calculations. The analysis considered the time limits required for an operator to terminate an inadvertent boron dilution in cold shutdown. A boron dilution event in cold shutdown is indicated to the operator by the boron dilution alarm logic in the Nuplex 80+ advanced control complex (ACC), as described in CESSAR-DC Section 7.7.1.1.10. This alarm logic is part of the control systems in the ACC not required for safety; see Section 7.7 of this report.

Reliance on the boron dilution alarm logic is taken in the safety analysis as the annunciator of the event and ensures that the 15-minute and 30-minute criteria discussed above are met. The alarms are redundant and available to enable the operator to detect and terminate an inadvertent boron dilution event within these required time intervals before shutdown margin is lost. Therefore, the intent of SRP Section 15.4.6 is met, and Issue 22 is resolved for the System 80+ design.

Issue 23: Reactor Coolant Pump Seal Failures

Issue 23, in NUREG-0933, addressed the concerns about RCP seal failures that could cause a SBLOCA. PRA analyses have indicated that the overall probability of core damage due to a small break could be dominated by RCP seal failures. This issue includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and under abnormal conditions. Specifically, acceptable resolutions to this issue include an RCP seal design that ensures the RCP seal integrity following SBO for an extended period.

In CESSAR-DC Section 20.2.5, ABB-CE states that the RCP seal for the System 80+ design is a ABB-CE KSB-designated seal similar to that used in the Palo Verde plant. The seal is cooled through two independent and redundant seal cooling systems: the seal injection system and the component cooling water system (CCWS). The seal injection system is one part of the CVCS, which receives power from a non-safety-related bus. The CCWS is a safety-grade system satisfying the single-failure criterion.

The System 80+ design also includes a control-grade onsite AAC power source to power the charging pumps that supply seal injection water to cool the RCP seal during an SBO. In response to staff RAI Q440.120 (listed in Appendix B of this report) regarding RCP seal integrity during an SBO for an extended period, ABB-CE evaluated the tests performed on RCP seal cartridge and loss-of-seal-cooling events that occurred at the Palo Verde plant. ABB-CE asserted that the RCP seal test data and plant data support the position that the RCP seal integrity will be maintained for an extended period. The seal cartridge tests were performed on Byron Jackson seals; but the System 80+ design uses ABB-CE KSB seal cartridges. ABB-CE claimed that these two types of seal cartridges are similar. However, no justification was given to show that the cartridges would behave identically. Therefore, the staff required that a supportable test be performed on the specific seal actually being used to demonstrate adequate RCP seal performance following total loss-of-seal cooling. Also, the plant data of the loss-of-seal-cooling events at Palo Verde contain significant uncertainties; therefore, the Palo Verde events cannot be considered as a definitive and conclusive test for seal integrity. The staff reviewed the May 14, 1992, submittal of ABB-CE, regarding tests of RCP seal performance for five different loss-of-cooling-flow conditions. Although ABB-CE stated that the RCP seal integrity is maintained for the test conditions, the staff finds that only Case 3 contains test conditions applicable to an SBO, and the Case 3 test was performed for only about a half hour during SBO conditions. The staff finds that this is not sufficient time to justify the seal integrity during an SBO for an extended period. The staff required that ABB-CE submit adequate test data to demonstrate the ABB-CE KSB seal integrity during an SBO for an extended period, or provide a diverse seal injection system, which should be independent of the CVCS and associated support systems to the extent practicable. This is DSER Open Item 20.2-7.

In its response dated July 29, 1993, and in CESSAR-DC Section 20.2.5, ABB-CE added a dedicated seal injection system (DSIS) as a diverse means of seal injection to the RCPs if normal means of seal cooling are lost. The DSIS air-cooled small-capacity positive displacement pump is placed in parallel with the centrifugal charging pumps in the CVCS and supplies the required 25 L/minute (6.6 gpm) flow through the normal CVCS seal injection path. As described in CESSAR-DC Section 9.3.4, the added positive displacement pump and it associated piping are part of the CVCS and are designed as ASME Code, Section III, Class 3. The DSIS can be aligned with the EDGs during a LOOP event, and an AAC power source in the event of an SBO. For a LOOP event, the EDGs will provide power to 2 of 4 available CCWS pumps to provide RCP seal injection. The charging pumps can also be powered by the EDGs to provide RCP seal injection. If CCWS and charging pumps are unavailable, the DSIS will be aligned to the EDGs to provide seal injection. During an SBO, AAC will be provided to a charging pump and CCWS pump (both with redundant availability) to provide seal cooling to the RCPs. If this means of seal cooling is lost, the DSIS will be aligned to the ACC power source and will be available within 10 minutes.

In order to ensure that the availability of the DSIS is maintained throughout the design, implementation, and operation phases, ABB-CE includes the DSIS in the design reliability assurance program (D-RAP) and the operations reliability assurance process (O-RAP). D-RAP and O-RAP are discussed in Section 17.3 of this report. Equipment included in D-RAP are listed in Table 17.3-4.

ABB-CE also incorporated the use of the DSIS into EOGs, which are discussed in Chapter 18 of this report. A requirement to maintain the availability of the DSIS was added to the instruction for meeting the RCP restartup criteria, which will be used in the following recovery guidelines: LOCA, SGTR, excess steam demand, loss of all feedwater, LOOP, and SBO. Step 5 of the LOOP recovery guidelines was modified to include the DSIS (in addition to CCWS and charging pumps) as one of the success paths in establishing the RCP seal cooling. Step 13 of the SBO recovery guidelines was also changed to ensure that vital ac power will be supplied to all the means of RCP seal cooling including the DSIS.

Since ABB-CE submitted a diverse and reliable DSIS design for RCP seal injection, and includes the DSIS in its D-RAP and O-RAP to ensure the high availability of the DSIS, the staff concludes that the design of DSIS in combination with the normal means (CCWS and charging pump seal injection) of RCP seal cooling will provide adequate RCP seal cooling and is acceptable. Therefore, DSER Open Item 20.2-7 is resolved.

Based on the above, Issue 23 is resolved for the System 80+ design.

Issue 24: Automatic ECCS Switchover To Recirculation

Issue 24, in NUREG-0933, addresses the concerns raised by the staff following a review of operating events that indicated a significant number of ECCS spurious actuations, particularly the four events that occurred at Davis Besse during 1980. Switchover from injection to recirculation involves realignment of several valves and may be achieved by (1) manual realignment, (2) automatic realignment, or (3) a combination of both. Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. In NUREG-0933, this issue was classified as medium-safety priority but has not been generically resolved.

The only source of water for ECCS for the System 80+ design is the in-containment water storage system, or IRWST, discussed in Section 6.8 of this report. As ABB-CE states in CESSAR-DC Section 20.2.6, there is no injection phase and recirculation phase for ECCS because water is always drawn, when needed for ECCS, from the IRWST and, therefore, there is no switchover for the System 80+ design for long-term cooling. This is discussed in Section 6.3.4 of this report.

Therefore, Issue 24 is resolved for the System 80+ design.

Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

Issue 29, in NUREG-0933, addressed staff concerns about the number of events involving the degradation of threaded fasteners (such as bolt cracking, corrosion, and failure) in operating plants from 1964 to the early 1980s. Many of the events were related to components of the RCPB and support structures of major components. This raised questions about the integrity of the RCPB and the reliability of the component support structures following a LOCA or a seismic event. Because licensees reported failures involving a variety of threaded fasteners and other causes, several different failure mechanisms had to be considered. Most frequent were wastage (corrosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints; the latter in structural bolting.

This issue was resolved and no new requirements were established based on (1) operating experience with bolting in both nuclear and conventional power plants; (2) actions already taken through bulletins, generic letters, and information notices since 1982; and (3) industry-proposed recommendations and actions, which are documented in EPRI Reports NP-5769 ("Degradation and Failure of Bolting in Nuclear Power Plants," EPRI, April 1988) and NP-5067 ("Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel," Volume 1: "Large Bolt Manual," 1987 and Volume 2: "Small Bolts and Threaded Fasteners," EPRI, 1990). The resolution of this issue is documented in GL 91-17, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants'," dated October 17, 1991; and NUREG-1339, "Resolution of Generic Safety Issue 29: 'Bolting Degradation or Failure in Nuclear Power Plants'," dated June 30, 1990.

ABB-CE indicates in CESSAR Section 20.2.7 that proven bolting designs, materials, and fabrication techniques will be employed. In addition, RCPB bolting will meet the requirements of ASME Code, Section III. Also, the owner/operator will observe established industry practice in developing maintenance, assembly, and disassembly procedures for RCPB bolting. Further, ISI will meet the requirements of ASME Code, Section XI for the RCPB and its support bolting.

In the DSER, the staff designated two open items on this issue. These DSER open items were resolved as follows:

(1) Although ABB-CE does not specifically reference GL 91-17, ABB-CE in Amendment L modified the CESSAR-DC to reference EPRI reports NP-5769 and NP-5067, as well as NUREG-1339. These documents, in addition to other industry-generated information, serve as the technical basis for the resolution of this issue by both ABB-CE and the owner/operator. Therefore, DSER Open Item 20.2-8 is resolved.

(2) While Issue 29 was being resolved, the staff addressed several specific issues on threaded fasteners in bulletins, generic letters, and information notices (e.g., PWR coolant pressure boundary bolting and component degradation due to boric acid corrosion; SCC of internal bolting in certain types of check valves; traceability and material control of fasteners; and nonconforming, misrepresented, counterfeit, and/or fraudulent bolting), the details of which are found in NUREG-1339. To address these, ABB-CE in Amendment L modified the CESSAR-DC to reference EPRI documents NP-5067 and NP-5769, which provide guidance for "design and construction" and "operation and maintenance" for fasteners of the RCPB. Additionally, ABB-CE has amended these references by also referencing NUREG-1339, which serves as the technical basis for the resolution of this issue by both ABB-CE and the owner/operator. Therefore, DSER Open Item 20.2-9 is resolved.

ABB-CE's proposal for Issue 29 is adequate in ensuring the structural integrity of bolting. Therefore, Issue 29 is resolved for the System 80+ design.

Issue 36: Loss of Service Water

Issue 36, in NUREG-0933, addressed the potential for the loss of both redundant trains of service water caused by the failure of a non-safety-grade system or component. This issue arose after Calvert Cliffs, Unit 1, experienced a loss of both redundant trains of service water when the station service water system (SSWS) became air bound as a result of the failure of a non-safety-related aftercooler in an instrument air compressor.

This issue was resolved with the issuance of SRP Sections 9.2.1 (Rev. 4) and 9.2.2 (Rev. 3). The revisions did not incorporate any new guidelines or requirements. The staff reviewed the System 80+ design to determine if it met the intent of these SRP sections. A detailed discussion of the staff's review is in Sections 9.2.1 and 9.2.2 of this report. Based on these sections, Issue 36 is resolved for the System 80+ design.

Issue 43: Reliability of Air Systems

Issue 43, in NUREG-0933, was initiated in response to an immediate action memorandum issued by the NRC Office for Analysis and Evaluation of Operation Data (AEOD) in September 1981 regarding desiccant contamination of instrument air lines. The memorandum was prompted by an incident at Rancho Seco where desiccant particles in the valve operator caused the slow closure of a containment isolation valve. Desiccant contamination in the instrument air system (IAS) was also found to be a contributing cause of the loss of the salt water cooling system at San Onofre in March 1980; this incident resulted in Issue 44, "Failure of the Saltwater Cooling System." Since the only new generic concern found in the evaluation of the San Onofre event was the common-cause failure of safety-related components due to contamination of the IAS, Issue 44 was combined with Issue 43.

Issue 43 was broadened to include all causes of air system unavailability because U.S. LWRs rely upon air systems to actuate or control safety-related equipment during normal operation even though they are not safety-grade systems at most operating plants. Safety system design criteria require (and plant accident analyses assume) that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform its intended function with the assistance of backup accumulators. The AEOD case study highlights 29 failures of safety-related systems that resulted from degraded or malfunctioning air systems. These failures contradict the requirement that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or will perform their intended function with the assistance of backup accumulators. Some of the systems that may be significantly degraded or failed are DHR, auxiliary feedwater, BWR scram, main steam isolation, salt water cooling, EDG, containment isolation, and the fuel pool seal system. The end result of degradation or failure of safety or safety-related systems is an increase in the expected frequency of core-melt events and, therefore, an increase in public risk.

This issue was resolved by the issuance of GL 88-14, "Instrument Air Supply Problems Affecting Safety-Related Equipment," dated August 8, 1988, which required licensees and applicants to review the recommendations of NUREG-1275 ("Operating Experience Feedback Report - Air Systems Problems," two volumes, dated July and December 1987, respectively) and perform a design and operations verification of the IAS. The following is a discussion of how ABB-CE considered the recommendations in NUREG-1275, Volume 2, for the System 80+ design:

(1) To ensure that air system quality is consistent with equipment specifications and is periodically monitored and tested.

In CESSAR-DC Section 9.3.1.2.1, ABB-CE states that to ensure the air system quality is consistent with equipment specifications, specifications meet the manufacturer's air supply requirements for all pneumatic equipment that is either safety-related or relied upon to perform a safety function.

(2) To ensure adequate operator response by formulating and implementing anticipated transient and system recovery procedures for loss-of-air events.

As described in CESSAR-DC Section 13.5.1.1, the COL applicant is responsible for developing and implementing anticipated transient and system recovery procedures for loss-of-air events.

(3) To improve training to ensure that plant operations and maintenance personnel are sensitized to the importance of air systems to common mode failures.

As described in CESSAR-DC Section 13.5.1.1, the COL applicant is responsible for developing and implementing plant operating and maintenance procedures for the IAS.

(4) To confirm the adequacy and reliability of safety-related backup accumulators.

The System 80+ design does not have any safety-related backup accumulators that are used to operate safety-related air-operated valves. However, safety-related accumulators are used in the diesel generator engine starting air system (CESSAR-DC Section 9.5.6). In CESSAR-DC Section 9.5.6.2.2, ABB-CD states that "a multi-stage drying and filtering unit . . . [is provided] to supply air with a dewpoint at least 10° [degrees]F lower than the lowest expected ambient temperature." Additionally, in Section 9.5.6.4, ABB-CE states that "System components and piping are tested to pressures designated by appropriate codes. Inspection and functional testing are performed prior to initial operation; thereafter, the system will be tested in accordance with the TS. Periodic blowdown of the starting air tanks is done to check for moisture. The frequency will be determined based upon operating experience. Air dryer desiccant is inspected per the manufacturers recommendations (approximately every six months). The COL applicant will make available for NRC review, information on Diesel Generator Engine Starting Air System test frequencies." This is COL Action Item 9.5.4.1-2.

(5) To verify equipment response to gradual losses of air to ensure that such losses do not result in events which fall outside FSAR analysis.

In CESSAR-DC Section 14.2.12.1.136, ABB-CE states, "Repeat Test A, but shut the instrument air system off very slowly to simulate a gradual loss of pressure." This is also addressed in CESSAR-DC Section 9.3.1.4 where it is stated that "The instrument air system preoperational testing and inspection is in accordance with the intent of RG 1.68.3 prior to initial operation."

Also, in CESSAR-DC Sections 9.3.1 and 20.2.9, ABB-CE states that the compressed air systems are not safety related and are, therefore, not needed during accidents to perform any safe shutdown or accident mitigation functions.

Based on the above, Issue 43 is resolved for the System 80+ design. The COL applicant will provide the information identified as COL Action Item 9.5.4.1-2 in Item (4) above.

Issue 45: Inoperability of Instruments Due to Extreme Cold Weather

Issue 45, in NUREG-0933, addressed the potential for safety-related equipment instrument lines to become inoperable as a result of freezing or reaching the precipitation point of the sensing fluids. Typical safety-related systems employ pressure and level sensors that use small-bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to ambient temperature conditions. These lines generally contain liquid (e.g., borated water) that is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to subfreezing temperatures, heat tracing or insulation or both is used to minimize the effects of cold temperatures. These sensing lines are of concern because, should they freeze, they may prevent a safety-related system or component from performing its safety function.

To resolve this issue, the staff issued RG 1.151, "Instrument Sensing Lines," to supplement the existing guidance and requirements in the SRP, applicable GDC, and standard ISA-67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." RG 1.151 addresses the prevention of freezing in safety-related instrument-sensing lines and includes such design issues as diversity, independence, monitoring, and alarms.

In CESSAR-DC Section 20.2.10, ABB-CE commits to the above documents and states that all safety-related systems and components used in the System 80+ design, including instrument-sensing lines, are located in a temperature-controlled environment that is maintained above the freezing (or precipitation) point of the contained fluid. Each building has a particular set of environmental control requirements that are maintained by heating, ventilation, and air conditioning (HVAC) systems designed for that specific task. LCOs for the ventilation systems that control the environment for the buildings that house the safety-related systems require that the plant be placed in a safe-shutdown condition should the temperatures in these buildings exceed specified ranges. This ensures that the safety-related systems and components are not exposed to freezing or other adverse conditions.

Locating all sensing lines inside environmentally controlled structures is an acceptable resolution of this issue. The staff's acceptance of the different System 80+ HVAC systems is discussed in Section 9.4 of this report.

Based on the above, Issue 45 is resolved for the System 80+ design.

Issue 48: Limiting Conditions for Operation for Class 1E Vital Instrument Buses in Operating Reactors

Issue 48, in NUREG-0933, addressed the concern that some operating nuclear power plants did not have TSs or administrative controls governing operational restrictions for Class 1E 120-V ac vital instrument buses and associated inverters. Without such restrictions, these power sources could be out of service indefinitely, thereby placing certain safety systems in a situation of not being able to meet the single-failure criterion.

In CESSAR-DC Section 20.2.11, ABB-CE states that it will identify the LCOs for onsite power systems, including the Class 1E 120-V ac vital instrument buses and associated inverters in CESSAR-DC Chapter 16 on plant TSs. The plant TSs will include specific requirements regarding plant operational restrictions as they apply to the Class 1E 120-V ac vital instrument buses and inverters. On this basis, the staff concludes that it will evaluate this aspect of the design when the proposed TSs were submitted for the System 80+ design. This was Open Item 20.2-10 in the DSER.

Subsequently, the staff has reviewed the System 80+ TSs and confirmed that the LCO for Class 1E vital instrument buses and associated inverters are included. Chapter 16 of the report discusses the System 80+ TSs. The staff finds ABB-CE's response to this issue acceptable. Therefore, DSER Open Item 20.2-10 is resolved.

In November 1986, this issue was integrated into Issue 128 on electric power reliability. The discussion on Issue 128 and the System 80+ design appears later in this section.

Therefore, Issue 48 is resolved for the System 80+ design.

Issue 49: Interlocks and LCOs for Class 1E Tie Breakers

Issue 49, in NUREG-0933, addressed the concern that tie breakers which can connect redundant safety-related buses require administrative controls to lock them open because, when closed, they can compromise the independence of the redundant Class 1E buses. The licensee's review of the electric power design of Point Beach Nuclear Plant, Units 1 and 2, identified that the design, under certain conditions, allowed manual interconnections of redundant electrical load groups, thereby paralleling their power sources, and it took the plant operators approximately five weeks to discover that the electrical distribution system lineup was not in the proper configuration. This also suggested a generic concern regarding the adequacy of procedural and administrative controls.

As stated in CESSAR-DC Section 20.2.12, the System 80+ design has no manual or automatic ties between the redundant Class 1E power systems. Also, double breakers maintain independence between the Class 1E and the permanent non-safety 4160 Vac buses. Therefore, the concern of Issue 49 does not exist in the System 80+ design.

In November 1986, this issue was also integrated into Issue 128 on electric power reliability. The discussion on Issue 128 and the System 80+ design appears later in this section.

Therefore, Issue 49 is resolved for the System 80+ design.

Issue 51: Improving the Reliability of Open-Cycle Service Water Systems

Issue 51, in NUREG-0933, addressed fouling of safety-related open-cycle service water systems by either mud, silt, corrosion products, or aquatic bivalves. This problem has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation in nuclear power plants. This issue was originally to address only aquatic bivalves. However, the issues on flow blockage in essential equipment caused by Corbicula (Issue 32) and service water system flow blockage caused by Blue Mussels (Issue 52) were incorporated into this issue, and Issue 51 was expanded to consider if the NRC staff should develop new requirements for improving the reliability of open cycle water systems. New requirements were issued in GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, on baseline fouling programs for nuclear power plants.

In CESSAR-DC Section 20.2.13, ABB-CE states that the System 80+ design SSWS and CCWS are described in CESSAR-DC Sections 9.2.1 and 9.2.2, respectively. The SSWS is designed to serve one NSSS, and each NSSS on a multi-unit site will have its own SSWS.

The System 80+ design features an SSWS that cools only the CCWS heat exchangers. Thus, the number of components and amount of piping that can become fouled is minimized (see CESSAR-DC Section 9.2.1.2). The CCWS is utilized as an intermediate system between the SSWS and the safety-related systems and components being cooled (see CESSAR-DC Figure 9.2.2-1). The CCWS is filled with demineralized water and treated with corrosion inhibitors. Water quality design features applicable to the CCWS are listed in CESSAR-DC Table 9.2.2-1.

The following are SSWS design features and interface requirements to minimize fouling of the CCWS heat exchangers and the SSWS piping, prevent flow blockage, and facilitate the maintenance of clean conditions:

- The SSWS pump structures must be equipped with safety-grade traveling screens with a screen wash system. The screen mesh size must prevent flow blockage of the pump inlets, and limit ingestion of biological fouling organisms and debris (see CESSAR-DC Section 9.2.1.2.1.4).

- Strainers are installed at the SSWS pump discharges. The strainers are the automatic backwash type, designed to retain particles consistent with the fouling design limits of the CCWS heat exchangers (see CESSAR-DC Section 9.2.1.2.1.5).

- When required by the site-specific water chemistry and environmental regulations, the ultimate heat sink water must be chemically treated to reduce organic and inorganic fouling, corrosion, and scaling, and to keep mud and silt in suspension (see CESSAR-DC Section 9.2.5.2).

- The station service water intake structure will be visually inspected once per refueling cycle for macroscopic biological fouling organisms, sediment, and corrosion. Inspections should be performed either by scuba divers or by dewatering the intake structure or by comparable methods. Any fouling accumulations will be removed (see CESSAR-DC Section 9.2.1.4.2).

  Also, samples of water and substrate will be collected annually to determine if biological fouling organisms have populated the water source. Upon the detection of biological fouling organisms, appropriate corrective action, such as the modification of the chemical treatment program, should be taken. However, consideration must be given to environmental regulations (see CESSAR-DC Section 9.2.5.4).

- The capability to clean SSWS surfaces is discussed in CESSAR-DC Section 9.2.1.3.

- The CCWS heat exchangers are either of the tube and shell or plate and frame design, dependent upon site selection (see CESSAR-DC Section 9.2.2.2.1.1). SSWS water flows through the tube side of CCWS shell and tube heat exchangers at a lower pressure than the CCWS shell to prevent contamination of the CCWS by in-leakage of SSWS water. In addition, the nominal flow conditions in CCWS heat exchanger tubes are in accordance with Heat Exchanger Institute standards for power plant heat exchangers.

- Adequate tube pull space is provided for periodic tube cleaning of the straight tube type of CCWS heat exchangers.

- The CCWS heat exchangers have a 15-percent thermal performance margin to allow for potential fouling between cleaning operations (see CESSAR-DC Section 9.2.2.2.1.1). The thermal performance can be verified using temporary instrumentation at test connections provided on each heat exchanger (see CESSAR-DC Sections 9.2.1.5 and 9.2.2.5).

- Wetted surfaces of the SSWS and CCWS are of materials selected on a site-specific basis to be compatible with the respective cooling water chemistries and water treatments. The guidelines used for the selection of CCWS heat exchanger tuber and tubesheet materials are given in CESSAR-DC Section 9.2.2.2.1.1.

- Sites at which ice could form on the ultimate heat sink are to be analyzed to show that the function of the ultimate heat sink will not be impaired during winter months. Where required, the intake structures must have a means of de-icing, such as warm-water recirculation, to prevent flow blockage at the SSWS pump inlets (see CESSAR-DC Section 9.2.5.1.3).

The staff will review site-specific aspects of the resolution of Issue 51 when it reviews the site-specific application from the COL applicant. Site-specific reviews will include maintenance and inspection program(s) and the determination of whether or not the COL applicant commits to meet all identified interface requirements of ABB-CE. This is included in COL Action Item 9.2.1-1.

Therefore, based on the above, the staff has determined that concerns in GL 89-13 have been adequately addressed and that Issue 51 is resolved for the System 80+ design.

Issue 57: Effects of Fire-Protection Systems Actuation on Safety-Related Equipment

Issue 57, in NUREG-0933, addresses fire-protection system (FPS) actuations that have caused adverse interactions with safety-related equipment at operating nuclear power plants. Experience shows that safety-related equipment subjected to water spray, as from the FPS, could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors. This issue has not been resolved and is classified in NUREG-0933 as a medium-safety priority.

As stated in CESSAR-DC Section 20.2.14, the System 80+ plant is designed to preclude water spray from the FPS onto safety-related equipment. The sprinkler systems protecting the safety-related equipment are of the automatic preaction sprinkler type. Actuation of these sprinkler systems requires opening the fusible link for individual sprinkler heads and detection of by-products of combustion, or heat, or both. In addition, the operator can isolate flow from the control room by isolating the subsphere building headers or, locally, by using manual isolation valves.

In order to prevent flood damage upon actuation of sprinkler systems, floor drains are provided and equipment is located to preclude the flooding of the equipment. In addition, in order to further reduce potential damage to safety-related equipment upon actuation of sprinkler systems, equipment is shielded and conduit ends are sealed where required, based on interaction reviews during detailed design and as built walkdowns.

The open ends of all vertical conduit, and the open ends of all horizontal conduit that terminate within 18 inches of a floor, will be sealed to keep water out.

It should not be necessary to shield equipment from the effects of water spray from overhead sprinkler systems since sprinklers in safety-related areas will be of the automatic pre-action type. Redundant safety-related equipment is separated with 3-hour-rated fire barriers that will confine the fire and fire-fighting operations to a single area. From a safe shutdown standpoint, it is assumed that the fire will render the equipment in the affected area inoperable, and safe shutdown will be accomplished using the redundant division. Therefore, the wetting of safety-related equipment in the affected area will be acceptable. All penetration seals in floors and walls up to a height of 61 cm (24 in.) will be waterproof to prevent water from the affected area from migrating to adjacent areas.

Safety-related equipment near fittings in the standpipe and interior fire hose system will be shielded as necessary to prevent damage from inadvertent discharge. Shielding locations will be determined following as-built walkdowns.

Inside the containment, where redundant division equipment is located in close proximity (i.e., within 6 m (20 ft) of each other), such as the motor-operated depressurization valves located at the pressurizer, shielding will be installed as deemed necessary following interaction review during as-built walkdowns.

In addition, as described in CESSAR-DC Section 9.5.1.8.3, detrimental effects to safety-related equipment due to discharge of fire protection water will be mitigated through the use of equipment-mounting pedestals, curbs, and floor drains of the proper size to accommodate the anticipated flow of water from the FPS.

The staff stated in the DSER that it would evaluate the resolution of this issue when it reviewed individual referencing applications. This was designated DSER COL Action Item 20.2-3. Based on further review, the staff now finds that ABB-CE's response to this issue is adequate for the System 80+ design without review of the referencing applications. Therefore, DSER COL Action Item 20.2-3 is resolved.

In CESSAR-DC Section 20.2.14, ABB-CE indicates that in order to reduce the potential damage to safety-related equipment due to actuation of sprinkler systems, equipment is shielded and conduit ends are sealed where required based on interaction reviews performed during detailed design and as-built walkdowns. Performance of the interaction reviews during detailed design and as-built walkdowns to determine the need for equipment shielding and conduit sealing are included in the COL Action Item 9.5.1-1 discussed in Section 9.5.1 of this report on fire protection.

On the basis of the above discussion, the staff concludes that ABB-CE has adequately addressed the safety concerns of Issue 57 in the CESSAR-DC and, therefore, Issue 57 is resolved for the System 80+ design.

Issue 64: Identification of Protection System Instrument-Sensing Lines

Issue 64, in NUREG-0933, addressed the establishment of guidance for the identification of the mechanical sensing lines connected to safety-related I&C systems. Sensing lines are an integral part of safety-related (protection) systems, and are essential to their reliable operation. Therefore, identification of these lines will facilitate the verification that these lines are appropriately separated and protected.

Industry has developed a standard for safety-related instrument-sensing lines in ISA-S67.02-1980, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," which includes identification criteria. As part of establishing its guidance for safety-related instrument-sensing lines, the NRC endorsed ISA-S67.02-1980 in RG 1.151.

The acceptance criteria for the resolution of this issue are that sensing lines that work together with safety-related I&C shall be identified in accordance with ISA-S67.02-1980 and shall meet the intent of the guidance in RG 1.151. Specifically, the instrumentation-sensing lines shall meet Section 5.3 of the ISA standard. Section 5.3, in part, states that the instrument-sensing lines related to safety-related instrumentation will be identified and color coded.

As stated in CESSAR-DC Section 20.2.15, ABB-CE states that the System 80+ design includes safety-related I&C that use mechanical sensing lines. These sensing lines will be identified and color coded in accordance with RG 1.151, to distinguish individual safety channels. In addition, the guidance in RGs 1.151 and 1.75, is imposed as design criteria for the routing of Class 1E (safety-related) and associated cabling and sensing lines from sensors. The safety-related I&Cs (including the sensing lines) meet the criteria in ISA-S67.02-1980, as invoked by the guidance in RG 1.151.

Therefore, Issue 64 is resolved for the System 80+ design.

Issue 66: Steam Generator Requirements

After the SGTR event at Ginna on January 25, 1982, the staff determined to develop generic SG requirements which would help mitigate or reduce SG tube degradations and ruptures. In September 1988, Issues A-3, A-4, and A-5, which addressed SG tube integrity, were resolved and the staff's findings were published in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," dated September 1988. The staff concluded that no new or revised requirements were needed because NUREG-0844 addressed the safety concerns identified under Issue 66. Before NUREG-0844 was finalized, the staff issued its recommendations on SG tube integrity from NUREG-0844 in GL 85-02. Because the staff used GL 85-02 to document recommendations, and not to add new requirements, this issue was resolved and no new requirements were established.

The resolution of this issue in terms of the implementation of GL 85-02 on the System 80+ design is also discussed in the resolution of Issue A-4 for the design in Section 20.2 of this report. The SG tube ISI program is discussed in the resolution of Issue A-4.

In CESSAR-DC Section 20.2.16, ABB-CE states that the secondary system, including the SGs and condenser, will be designed, manufactured, tested, inspected, and operated in accordance with accepted industry codes and standards. The SGs will meet the requirements of Sections III and XI of the ASME Code for design, manufacture, test, and inspection. Also, the SG design will meet the intent of the guidance given in SRP Section 5.4.2.1, "Steam Generator Materials," for SG materials. The staff evaluated the SGs against the SRP section in Section 5.4.2 of this report.

Therefore, ABB-CE's statements are adequate in ensuring the structural integrity of SG tubes and Issue 66 is resolved for the System 80+ design.

Issue 67.3.3: Improved Accident Monitoring

Issue 67.3.3, in NUREG-0933, addressed weaknesses in reactor system monitoring that could inhibit correct operator responses to events similar to the SGTR event at the Ginna Power Plant on January 25, 1982. During the event, the following weaknesses in accident monitoring were apparent: (1) non-redundant monitoring of RCS pressure, (2) failure of the position indication for the SG relief and safety valves, and (3) limited range of the charging pump flow indicator. As stated in NUREG-0933 and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980, (Supplement 1, January 1983), the implementation of the recommendations described in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," resolves this issue.

In the DSER, the staff reviewed the safety-related plant process display instrumentation information presented in CESSAR-DC Section 7.5 using the guidelines of RG 1.97. Except for Type A variables, as explained in RG 1.97, the information ABB-CE presented in the initial CESSAR-DC Section 7.5 conformed to the guidelines of RG 1.97. Type A variables provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBEs. The initial discussion of ABB-CE about Type A variables was not adequate for the staff to conclude that the System 80+ accident monitoring capabilities were acceptable. Resolution of this issue was designated as DSER Open Item 7.5.2.1-1.

As discussed in Section 7.5 of this report, ABB-CE stated that all protective functions in the System 80+ design use automatic or passive responses, such that operator action is not required to accomplish the protective function. Therefore, no Class 1E alarms and no Type A variables are required. The staff finds acceptable the selection from ABB-CE of indications for monitoring post-accident conditions. The listing of these variables satisfies the requirements of 10 CFR 50.34(f)(2)(xvii) (TMI Action Plan Item II.F.1) and 10 CFR 50.34(f)(2)(xix) (TMI Action Plan Item II.F.3). Both Issues II.F.1 and II.F.3 are discussed in Section 20.4 of this chapter. This resolves DSER Open Item 7.5.2.1-1.

Therefore, based on the above, the staff finds that ABB-CE has acceptably addressed Issue 67.3.3, and it is resolved for the System 80+ design.

Issue 70: PORV and Block Valve Reliability

Power-operated relief valves (PORVs) and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation; the block valves were installed because of expected leakage from the PORVs. Neither valve type was needed to safely shut down a plant or mitigate the consequences of accidents. In 1983, the staff determined that PORVs were relied on to mitigate design-basis SGTR accidents and questioned the acceptability of relying on non-safety-grade components to mitigate DBAs. Issue 70, in NUREG-0933, addressed the assessment of the need for improving the reliability of PORVs and block valves.

In CESSAR-DC Section 20.2.18, ABB-CE states that the design purpose of PORVs is to prevent challenges to spring-operated safety valves and to provide rapid depressurization capability. Older ABB-CE plants had PORVs, but the newer ABB-CE System 80+ design does not include PORVs and block valves. Instead, the System 80+ includes a SDS, which is a safety-grade system, providing venting and rapid depressurization capability for mitigation of beyond-design-basis accidents. In Section 6.7 of this report, the staff approved the design of the SDS.

Therefore, Issue 70 is resolved for the System 80+ design.

Issue 75: Generic Implications of ATWS Events at Salem Nuclear Plant

Issue 75, in NUREG-0933, addressed the generic implications of two events at Salem Unit 1 where there were failures to scram automatically because of the failure of both reactor trip breakers to open on receipt of an actuation signal. This issue was expanded to include a number of issues raised by the staff that were closely related to the design and testing of the RPS. The requirements for this issue were stated in GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event," dated July 8, 1983.

The actions covered by GL 83-28 fall into the following four areas:

(1) Post-Trip Review — This action addresses the program, procedures, and data collection capability to ensure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

The System 80+ DPS records and displays all system parameters for subsequent use by plant personnel. The staff evaluated the DPS and found the DPS to be acceptable. The adequacy of the program and procedures for post trip reviews will be determined during the staff's review of the COL applicants' practices.

(2) Equipment Classification and Vendor Interface — This action addresses the programs for ensuring that all components necessary for performing required safety-related functions are properly identified in documents, procedures, and information-handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

The adequacy of the equipment classification and vendor interface will be determined during the staff's review of the COL applicants' practices.

(3) Post-Maintenance Testing — This action addresses post-maintenance operability testing of safety-related components.

The adequacy of post-maintenance testing will be determined during the staff's review of the COL applicant's procedures.

(4) RTS Reliability Improvements — This intent of this action is to ensure that (a) vendor-recommended reactor trip breaker modifications and associated RPS changes are completed in PWRs, (b) that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, (c) that the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their RTS, and (d) to ensure that on-line functional testing of the RTS is performed on all LWRs.

The adequacy of RTS reliability improvements will be determined during staff's review of the COL applicant's procedures. On-line function testing of the RTS is addressed in the resolution of Issue 120 later in this section.

The staff finds the RPS design and qualification for the System 80+ design to be acceptable, as discussed in Section 7.2 of this report.

ABB-CE discussed the System 80+ design in terms of ATWS in CESSAR-DC Section 20.2.53. The staff evaluated the System 80+ ATWS system and found it acceptable as described in the discussion of Issue A-9 in Section 20.2 of this chapter. The actions covered by GL 83-28 in the above four areas will be provided by the COL applicant, as part of COL Action Item 13.5-1 which is discussed in Section 13.5 of this report. The staff concludes that the System 80+ reactor trip design satisfies the criteria of GL 83-28.

Therefore, Issue 75 is resolved for the System 80+ design.

Issue 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System

Issue 78, in NUREG-0933, addressed the fact that repeated thermal cycling of RCS components produces some degree of fatigue degradation of the material that could lead to failure, thereby increasing the likelihood of a LOCA. There was concern that there are no TS requirements for monitoring the actual number of transient occurrences for many older operating reactors. Licensees of newer operating reactor are required to keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. Additionally, the staff determined that the fatigue curves used in ASME Code, Section III may not be adequate to account for environmental effects. Data indicated that the existing code fatigue curves may have less margin than originally intended when considering the effects of fatigue induced by the operating environment.

A possible solution to Issue 78 identified in NUREG-0933 was to require affected plants to monitor transients and to verify that the design life of all ASME Code, Section III, Class 1 components have not been exceeded. Plants that have experienced transient events that exceeded design limits would perform fatigue analyses to determine the number of remaining thermal cycles before fatigue limits are exceeded, including consideration of environmental effects on the fatigue life. For ASME Code, Class 2 and 3 components that are subjected to cyclic loading, an appropriate analysis would be required.

In CESSAR-DC Section 20.2.20, ABB-CE states that plant transients that need to be considered in the design and fatigue analyses for System 80+ components are described in CESSAR-DC Section 3.9.1. Transients expected over a 60-year plant life are listed in CESSAR-DC Table 3.9-1. Fatigue monitoring for Class 1 components will be performed by the owner/operator in accordance with Section 5.7.2.9 of the System 80+ TSs as described in CESSAR-DC Section 16.15.7. The environmental effects of fatigue are discussed in CESSAR-DC Section 3.9.1.1. In accordance with CESSAR-DC Section 3.9.3.1.3, Class 2 and 3 components are reviewed for thermal fatigue effects using the ASME Code, Section III, NC-3219-2 for guidance. For components not meeting NC-3219-2 criteria, fatigue analysis is performed in accordance with NC-3200.

The staff's evaluation of the information provided by ABB-CE is in Sections 3.9.2, 3.12.5.2, 3.12.5.7, and 3.12.5.8 of this report. The staff concludes that ABB-CE's proposed resolution of this issue is acceptable.

Therefore, Issue 78 is resolved for the System 80+ design.

Issue 79: Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Issue 79, in NUREG-0933, addressed the concern for an unanalyzed reactor vessel thermal stress during natural convection cooldown (NCC) of PWR reactors. The concern emerged from a preliminary evaluation of the voiding event that occurred in the upper head of the St. Lucie Unit 1 reactor on June 11, 1980. On the basis of several conservative assumptions, Babcock and Wilcox tentatively concluded that during natural convection cooling, axial temperature gradients could develop in the vessel flange area which could produce thermal stresses in the flange area or in the studs that might exceed values allowed by the code when added to the stresses already considered (such as boltup loads or pressure loads).

This issue was resolved and no new requirements were established because (1) NCC events that result in the plant being brought to a cold shutdown condition occur infrequently and (2) the actual severity of a specific NCC event will determine the need for actions (if any) and the extent of actions that may be required of any licensee following certain NCC events that may place a reactor vessel in an unanalyzed condition or outside its documented design basis. The resolution of Issue 79 is documented in GL 92-02, "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown'," and NUREG-1374, "An Evaluation of PWR Reactor Vessel Thermal Stress During Natural Convection Cooldown," dated May 1991. There were no new design requirements in GL 92-02.

ABB-CE states in CESSAR-DC Section 20.2.21 that the design of the reactor pressure vessel (including the head and studs) will accommodate the thermal stresses caused by an NCC event. These thermal stresses, when added to stresses from events that are presently analyzed, will not exceed the stress limits specified in the ASME Code, Section III.

ABB-CE also states that stress analyses were performed to determine the effects of a natural circulation cooldown event on both the St. Lucie "class" reactor vessel and the System 80 "class" reactor vessel. It was concluded that should natural circulation cooldown of the RCS be required and should vessel head voiding subsequently occur, the resulting thermal stresses would not cause any thermal, hydraulic, or fatigue damage to the reactor vessel and its integral components over their design lifetime.

The System 80+ reactor vessel, which is designed to the ASME Code, Section III, is essentially identical to the System 80 reactor vessel. Because the reactor vessels for both "classes" of plants are virtually the same, and since the stress analyses consider the materials, dimensions, and geometry of the vessel, ABB-CE contends that the analyses performed subsequent to the St. Lucie event also apply to the System 80+ reactor vessel. By DSER Open Item 20.2-11, the staff requested that ABB-CE submit additional information regarding the applicability of the St. Lucie and System 80 analyses to System 80+. ABB-CE responded in that CESSAR-DC Tables 1.3-1, 3.9-17 and 3.9-18, and Figure 3.9-9 provide a detailed comparison of the design, geometry, and operational parameters between System 80 and System 80+, and show that the two reactor vessels were essentially identical.

The staff also requested in the DSER that ABB-CE verify the number of NCC events for the lifetime of the System 80+ design. ABB-CE states in CESSAR-DC Table 3.9-1 that 30 NCC events are included in the design-bases events for thermal, hydraulic, and fatigue analyses for System 80+. In NUREG-0933, NCC events are estimated to occur about 0.04 times per year, which would be about 3 times in 60 years, and, thus, the commitment of ABB-CE to include 30 NCC events in the System 80+ design is acceptable. Therefore, DSER Open Item 20.2-11 is resolved.

ABB-CE's proposed resolution of Issue 79 adequately ensures the structural integrity of the reactor vessel and, therefore, Issue 79 is resolved for the System 80+ design.

Issue 82: Beyond-Design-Basis Accidents in Spent Fuel Pools

The risks of beyond-design-basis accidents in the spent fuel storage pool were examined in WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975, and it was concluded in the report that these risks were orders of magnitude below those involving the reactor core. Issue 82 in NUREG-0993 was the reexamination of accidents in the spent fuel storage pool. The reasons are two-fold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high-density-storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have offered evidence of the possibility of fire propagation between assemblies in an air-cooled environment. These two reasons, in combination, provide the basis for an accident scenario that was not previously considered.

As stated in NUREG-0933, because of the large inherent safety margins in the design and construction of spent fuel pools, this issue was resolved and no new requirements were established.

The spent fuel pool and the storage racks for the System 80+ design are described in CESSAR-DC Section 9.1.2, the spent fuel pool cooling and cleanup system is described in Section 9.1.3, and the fuel-handling system (which includes the equipment for handling heavy loads) is described in Section 9.1.4.

In the DSER, the staff stated that pending resolution of the open items and confirmatory items identified in Sections 9.1.2, 9.1.3, and 9.1.4 of the DSER, ABB-CE's resolution for Issue 82 was acceptable. All open and confirmatory items identified in DSER Sections 9.1.2, 9.1.3, and 9.1.4 have been resolved as described in Section 9.1 of this report.

Therefore, Issue 82 is resolved for the System 80+ design. See also the resolution of Issue A-36 in Section 20.2 of this chapter.

Issue 83: Control Room Habitability

Issue 83, in NUREG-0933, addressed the significant discrepancies found during a survey of existing plant control rooms before 1983. These discrepancies included the inconsistencies between the design, construction, and operation of the control room habitability systems and the descriptions in the licensing-basis documentation. In addition, the staff determined that total system testing was inadequate and that the control systems were not always tested in accordance with the plant TS. The following issues are related to Issue 83: (1) Issue B-36 on criteria for air filtration and adsorption units for atmospheric cleanup systems, (2) Issue B-66 on control room infiltration measurements, and (3) Issue III.D.3.4 also on control room habitability. These three issues are discussed elsewhere in Sections 20.2 and 20.4 of this chapter.

The System 80+ design main control room habitability system is described in CESSAR-DC Sections 6.4 and 9.4.1, and the design bases are given in CESSAR-DC Section 6.4.1. The staff's evaluation is in Sections 6.4 and 9.4.1 of this report. Also, the HVAC systems for smoke removal from specific areas as a means of satisfying the smoke control provisions of Fire Protection Association

(NFPA) Guideline 90A are evaluated in Section 9.5.1 of this report.

The control room is a structure that is important to safety and, as such, is designed to withstand the effects of natural phenomena (such as earthquakes and hurricanes) and postulated accidents and missiles. The design is, therefore, specifically in accordance with GDC 2 and 4 (see CESSAR-DC Sections 3.1.2 and 3.1.4). Although the System 80+ design can be used at either single-unit or multiple-unit sites, ABB-CE stated in CESSAR-DC Section 1.2.1.3 that the independence of safety-related systems and their support systems will be maintained between (or among) the individual plants.

In Section 6.4 of the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant has to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. Upon further review, as discussed in Section 6.4 of this report, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the System 80+ design conforms to the requirements of GDC 5. In addition, the design of the control room permits safe occupancy during abnormal conditions and meets the requirements of GDC 19 (see CESSAR-DC Section 3.1.15).

The control room ventilation and air conditioning systems are designed for uninterrupted safe occupancy of the control room during postaccident shutdown in accordance with GDC 2, 4, 5, 19, and 60 (see CESSAR-DC Section 9.4.1.1). The control room is protected from fire by alarm systems and portable fire extinguishers (see CESSAR-DC Section 6.4.1). The testing requirements for the habitability system are in CESSAR-DC Sections 6.4.4 and 9.4.1.4.

In the DSER, the staff stated that the COL applicant would ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators were protected in accordance with TMI Action Plan Item III.D.3.4. In specific terms, the COL applicant would verify that the following are consistent with the licensing-basis documentation

- as-built design
- operating, maintenance, and emergency procedures
- training
- performance characteristics of the control room habitability system
- TSs and surveillance procedures

The COL applicant would also submit adequate verification of system performance and integrity. These concerns were identified in the DSER as COL Action Item 20.2-4. As discussed in Sections 6.4 and 9.4.1 of this report, the staff has concluded that the control room habitability systems and the control complex ventilation system comply with GDC 4, 5 and 19 by conforming to the acceptance criteria of SRP Sections 6.4 and 9.4.1. This adequately addresses the concerns identified in DSER COL Action Item 20.2-4 and DSER COL Action Item 20.2-4 is resolved.

On the basis of this discussion, the staff concludes that ABB-CE has adequately addressed the safety concerns of Issue 83 in the CESSAR-DC. Therefore, Issue 83 is resolved for the System 80+ design.

Issue 84: ABB-CE PORVs

Issue 84 of NUREG-0933 addressed concerns about ABB-CE PORVs. The design purpose of PORVs is to prevent challenges to spring-operated safety valves and to provide rapid depressurization capability. Older ABB-CE plants had PORVs, but the newer ABB-CE System 80+ design does not include PORVs.

As specified in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 13, 1990, the acceptance criterion for this issue is that the advanced reactor design shall include a safety-grade RCS safety depressurization and vent system for mitigation of beyond-design-basis events.

ABB-CE states in CESSAR-DC Section 6.7 that the System 80+ design includes the rapid depressurization system (RDS), which is a safety-grade system, providing venting and rapid depressurization capability for mitigation of beyond-design-basis accidents. In Section 6.7 of this report, the staff approved the design of the RDS.

Therefore, Issue 84 is resolved for the System 80+ design.

Issue 87: Failure of High-Pressure Coolant Injection Steamline Without Isolation

Issue 87, in NUREG-0933, addressed the staff concerns about a postulated break in the high-pressure coolant injection (HPCI) steam supply line and the uncertainty regarding the operability of the isolation valves for the HPCI steam supply line under these conditions. For the System 80+ design, this system is the SIS. The operation of these valves is tested periodically without steam and, due to flow limitations at the valve manufacturers' facilities, only the operating characteristics under operating conditions may be tested. Therefore, the capability of the valves to close when exposed to the forces created by the flow resulting from a break downstream may not be demonstrated. This issue was resolved by the issuance of GL 89-10, "Safety-Related Motor Operated Valve (MOV) Testing and Surveillance," dated June 28, 1989, and its supplements on safety-related MOV testing.

In Section 6.3 of this report, the staff discusses the SIS for the System 80+ design. Steamlines are not used to power the four high-pressure safety-injection pumps, and there are no low-pressure injection pumps. ABB-CE states this in CESSAR-DC Section 20.2.24. The lack of HPCI steamlines resolves this issue for the System 80+ design.

Issue 93: Steam Binding of Auxiliary Feedwater Pumps

Issue 93, in NUREG-0933, addressed the potential for a common-mode failure of the auxiliary feedwater or EFWS resulting from steam binding of the EFW pumps caused by heated MFW leaking back through check valves. The EFWS is used in the System 80+ design to supply water to the SGs should the MFW system be lost, and steam binding of the EFW pumps could result in the loss of the EFWS.

The EFWS may be isolated from the MFW system by a check valve or one or more isolation valves (depending upon the specific design) to keep hot MFW from entering the EFWS. However, operating experience has shown that check valves tend to leak, thus, permitting hot MFW to enter the EFWS. This hot feedwater can subsequently flash to steam in the EFW pumps and discharge lines causing steam binding of the pumps.

In addition, the EFW piping is sometimes arranged so that each EFW pump is connected through a single check valve (which is used to prevent back leakage) to piping that is common to two or three pumps. This arrangement creates the potential for common-mode failures as the hot feedwater leaks back through the check valves into other EFW pump(s).

The staff issued GL 88-03 ("Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps," dated February 17, 1988) to the industry as the resolution of this issue. The letter implements monitoring and corrective procedures to minimize the likelihood of steam binding of the EFWS pumps. One of the corrective actions to be taken is the monitoring of EFW pump discharge piping temperatures to ensure that the fluid temperatures remain at or near ambient temperature.

In CESSAR-DC Section 20.2.25, ABB-CE states that the EFWS in the System 80+ design includes two independent trains, each train aligned to supply its respective SG.

The System 80+ design is configured as two separate mechanical systems. Each mechanical system which consists of two subtrains (see CESSAR-DC Section 10.4.9 and the staff's evaluation in Section 10.4.9 of this report), contains (1) one emergency feedwater storage tank (EFWST), (2) a motor-driven and a steam-driven pump [each with a capacity of 1900 L/min (500 gpm)], (3) a cavitating Venturi, and (4) specified instrumentation. Each subtrain contains (1) one flow control valve, (2) one SG isolation valve, and (3) one pump discharge check valve.

The main defense against steam binding of the EFW pumps is the system design for normal plant operation.

Although some plant systems operate with the flow control and the isolation valves open during normal plant operations, the System 80+ EFWS is designed to operate with the EFW SG isolation valves closed for both mechanical systems. The isolation valves close in series with the SG line check valves, thus, providing redundant isolation between the EFWS and the MFW system. When a low level occurs in a SG, the EFAS or APS starts the EFW pumps (the motor-driven and steam-driven pumps), opens these isolation valves, and ensures that the feedwater flow control valves are open, allowing EFW to flow to each SG.

Each EFW subtrain is separated from the other. Each subtrain has its own suction line from the EFWST and its own discharge line through the SG isolation valve and check valve; also, the pump crossover lines contain redundant, locked-closed isolation valves. Thus, the potential for common-mode failure of steam binding of all EFW pumps does not exist, should one set of SG isolation and check valves leak. The EFW pump suction and recirculation lines are normally open so that, should a SG isolation and check valve leak, any resulting steam can be vented through the EFWST vent.

Associated instrumentation on each train ensures adequate control and monitoring of the EFWS. Temperature indicators (TIs) are located between the flow control and motor-operated isolation valves. These TIs give a direct indication of the fluid temperature and will alarm in the control room on high fluid temperature in the EFWS downstream of the EFW pumps.

This alarm warns the operator that the SG isolation valve and check valve are leaking. Therefore, these sensors provide an indication to the operator of the potential for steam binding of the EFW pumps.

The temperature sensor located between the EFW flow control valve and the isolation valve on each subtrain is continuously monitored and audibly alarmed in the control room.

In the event of loss of control room indication, the sensor will be monitored locally. At a minimum, readings shall be recorded at least once a shift, and before and after each EFW pump run. The site-specific surveillance and operating procedures, which are the responsibility of the owner/operator, will present the requirements for local monitoring.

The EFWS is designed to avoid steam binding of the EFW pumps by continuous system venting through the EFW storage tanks and by the use of normally closed isolation valves upstream of where the EFWS joins the MFW system; however, in the event that steam binding of the EFW pumps does occur, the control room alarm associated with the temperature sensor discussed above will signal the plant operator to vent the EFW pumps. Plant operating procedures developed by the owner/operator will prescribe this action.

The staff stated that it would evaluate the procedures for preventing or coping with steam binding of the EFW pumps on a site-specific basis during a COL review. This is discussed in Section 10.4.9 of this report and is designated COL Action Item 10.4.9-3.

On the basis of this discussion, the staff concludes that ABB-CE has adequately addressed the safety concerns of Issue 93 in the CESSAR-DC and Issue 93 is resolved for the System 80+ design.

Issue 94: Additional Low-Temperature Overpressure Protection for LWRs

Issue 94, in NUREG-0933, addressed low-pressure overpressurization events since the resolution of Issue A-26, which is also discussed in this chapter. Therefore, this issue was to address the additional guidance for RCS low-temperature overpressure protection (LTOP) to ensure reactor vessel integrity beyond the requirements specified for Issue A-26 in SRP Section 5.2.2, "Overpressure Protection," and BTP RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperature." Issue 94 was resolved with the additional requirements to have the TSs for overpressure protection consistent with those specified in Enclosure B to GL 90-06, "Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors, Pursuant to 10 CFR 50.54(f)," dated June 25, 1990.

In CESSAR-DC Sections 5.2.2, 5.4.10, 5.4.13, and Appendix 5A, ABB-CE describes the LTOP design and its compliance with the requirements of SRP Section 5.2.2 and BTP RSB 5-2. The staff has reviewed and approved, in Section 5.2.2 of this report, the LTOP design for the System 80+ plant. The TSs for LTOP are in CESSAR-DC Sections 16.7.11 and 16A.7-11. The LTOP function is performed by the relief valves in the SCS. The analysis to support the adequacy of the LTOP is in CESSAR-DC Section 5.2.2 and is approved as indicated in Section 5.2.2 of this report.

In the DSER, ABB-CE was requested by the staff, in accordance with the requirement of Enclosure B to GL 90-06, to revise the TSs for LTOP by reducing allowable outage time for a single channel from 7 days to 24 hours when the plant is operating in Modes 5 or 6. This was designated DSER Open Item 20.2-12. In response, ABB-CE revised surveillance requirements specified in the TSs in CESSAR-DC Section 16.7.11, to be consistent with the requirements in Enclosure B to GL 90-06. Therefore, DSER Open Item 20.2-12 is resolved.

ABB-CE has provided an acceptable LTOP design and the associated TSs are consistent with the requirements of BTP RSB 5-2 and Enclosure B to GL 90-06; Therefore, Issue 94 is resolved for the System 80+ design.

Issue 99: RCS/RHR Suction Line Valve Interlock on PWRs

Issue 99, in NUREG-0933, addressed the staff's concerns about the inadvertent closing of the SCS suction valves when the SCS is in use. In existing plants, the autoclosure interlocks (ACIs) on suction isolation valves of the SCS were installed to guard against an operator error, namely, failure to isolate the SCS from the RCS before raising the RCS pressure above the design pressure of the SCS; however, ACIs have been a frequent cause of loss-of-SCS events. As the result of a regulatory analysis in NUREG-0933, the staff recommended removing ACIs to improve the reliability of the SCS. Over the past several years, efforts have increased to improve the reliability of the SCS. In 1988, the staff broadened the scope of this issue to include risk associated with midloop operation. In GL 88-17, "Loss of Decay Heat Removal," dated October 17, 1988, the staff asked licensees to address numerous safety concerns and to improve the operational safety for such plant conditions as reduced RCS water inventory.

ABB-CE discussed this issue in CESSAR-DC Section 20.2.27. To address the concerns of this issue regarding the reliability of the SCS, the System 80+ design contains the following instrumentation consistent with the guidelines in GL 88-17:

- two independent level indications for continuous management of RCS level
- two independent channels of core exit thermocouples (CETs) for continuous measurement of core exit temperature
- indication and alarms in the control room for each of the level and temperature measurement instruments given in the preceding two items with each instrument of an independent pair having its own separate power supply
- additional instrumentation, indication, and alarms for plant parameters (such as SCS pump suction pressure and motor current, vortex monitoring equipment, and SCS flow) enabling operators in the control room to continuously monitor the SCS when it is in use for RCS cooling
- no ACI on the SCS suction isolation valves

The SCS design is consistent with the guidelines for removal of ACIs. Additional improvements are discussed in Section 5.4.3.2 of this report. Also, the guidelines in GL 88-17 cover broader areas, such as instrumentation, equipment, procedures, analyses, TS, and RCS perturbation for operation at low RCS water inventory.

Additionally, on a related RHR performance issue, the NRC staff has been concerned about the safety of operations during low power or periods of shutdown. The Diablo Canyon event of April 10, 1987, and the loss of ac power at the Vogtle plant on March 20, 1990, have led the staff to issue NUREG-1410, "Loss of Vital AC Power and Residual Heat Removal System During Midloop Operations at Vogtle Unit 1 March 20, 1990," dated June 1990, and to conclude that (1) non-routine activities and availability of less equipment during shutdown increases the probability of complex events that challenge operators in unfamiliar ways and (2) lack of rigorous consideration of accident sequences during shutdown operations has resulted in potentially incomplete or inadequate instrumentation, emergency response procedures, and mitigative equipment. Owing to the safety significance of events during shutdown and low-power conditions, the staff determined that proper consideration of the topic would be required before NRC would issue a final design approval on the System 80+ design.

Two primary measures were required to demonstrate adequate treatment of shutdown risk for the System 80+ design: (1) adequate vendor assessment of shutdown and low-power risk, identifying design-specific vulnerability and weakness and (2) documentation showing consideration and incorporation of design features that minimize shutdown and low-power risk vulnerabilities.

In response to staff RAI Q440-129 through 151, listed in Appendix B of this report, regarding the shutdown risk concerns, ABB-CE submitted a report on July 31, 1992. The report was incorporated in CESSAR-DC Section 19.8A and covered the following topics:

- procedures
- TS improvement
- midloop operation
- loss of DHR
- primary/secondary containment capability and source term
- rapid boron dilution
- fire protection
- instrumentation
- ECCS recirculation capability
- effect of PWR upper internals
- fuel handling and heavy loads
- potential for draining the reactor vessel
- CESSAR-DC Chapter 15: non-LOCA/LOCA dose consequences
- CESSAR-DC Chapter 6: LOCAs
- CESSAR-DC containment analysis
- PRA

In September 1993, the staff published NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States." This report addresses, among other things, proposed regulatory requirements for shutdown and low-power operations. Previously, the staff reviewed the shutdown risk evaluation report from ABB-CE against the guidance in NUREG-1449 (DSER Open Item 20.2-13). This is discussed in Section 19.3.8 of this report. The staff concludes that the shutdown risk evaluation of ABB-CE is acceptable and adequately addresses the requirements in NUREG-1449.

Therefore, based on the above, Issue 99 is resolved for the System 80+ design and DSER Open Item 20.2-13 is resolved. The staff's overall evaluation of the shutdown risk of the System 80+ design is in Section 19.3 of this report.

Issue 103: Design for Probable Maximum Precipitation

Issue 103, in NUREG-0933, addressed the acceptable methodology for determining the design flood level for a particular plant site. The use of the most recent National Oceanic and Atmospheric Administration (NOAA) procedures for determining the probable maximum precipitation for a site was questioned after a licensee disputed the use of two of NOAA's hydrometeorological reports. The issue was resolved with the revisions to SRP Sections 2.4.2 and 2.4.3 in 1989 to incorporate the probable maximum precipitation procedures and criteria contained in the latest National Weather Service publications. This was documented in the Federal Register Notice 54 FR 31268 on July 27, 1989, and GL 89-22, "Potential for Increased Roof and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service," dated October 19, 1989.

In CESSAR-DC Section 20.2.28, ABB-CE states that site design parameters, including maximum flood level, are given in CESSAR-DC Table 2.0-1. ABB-CE also states that the System 80+ plant is designed in accordance with GDC 2 for the most severe environmental conditions, such as flooding, tornado, and hurricane, and meets the intent of SRP Sections 2.4.2, "Flood," and 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers." Furthermore, the COL applicant will review historical site-specific environmental data to ensure compliance with the enveloping assumptions of CESSAR-DC Table 2.0-1. This is COL Action Item 2.0-1 of CESSAR-DC Table 1.10-1.

In CESSAR-DC Table 2.0-1, ABB-CE specifies the site-specific maximum flood level to be 0.3 m (1 ft) below grade. This flood level is acceptable because the minimum design-basis flood level is specified at 0.3 m (1 ft) below plant grade for preventing damages to seismic Category I SSCs as discussed in Sections 2.6 and 3.4 of this report. Any sites with a flood level higher than 0.3 m (1 ft) below grade will be excluded from the System 80+ design certification.

The COL applicant must use site-specific environmental data for determining probable maximum precipitation in accordance with the guidance of SRP Sections 2.4.2 and 2.4.3. This is to ensure the maximum flood level for the System 80+ design specified in CESSAR-DC Table 2.0-1 shall not be exceeded by the site-specific flood level. This is discussed in Section 2.4 of this report and is included in COL Action Item 2.4-1.

Therefore, based on this information, Issue 103 is resolved for the System 80+ design.

Issue 105: Interfacing System LOCA (ISLOCA) at LWRs

Issue 105, in NUREG-0939, was limited to PIVs in BWRs and was resolved by requiring leak-testing of the check valves that isolate low-pressure systems that are connected at the RCS outside of containment. It is related to Issue 96 which addressed PIVs between the RCS and RHR systems in PWRs. As stated in NUREG-0933, the staff issued Information Notice (IN) 92-36, Intersystem LOCA Outside Containment," dated May 7, 1992, on this subject. The individual plant examinations required by the staff on operating plants included analyses of these sequences. This issue was resolved without any new requirements.

The staff position regarding ISLOCA protection, as stated in SECY-90-016, is that future ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure. Note that the degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packings, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The design should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

For all interfacing systems and components that do not meet the full RCS URS criteria, ABB-CE needed to justify why it is not practicable to reduce the pressure challenge any further and to provide compensating isolation capability. For example, applicants should demonstrate for each interface that the degree and quality of isolation or reduced severity of the potential pressure challenges compensate for and justify the safety of the low-pressure interfacing system or component. The adequacy of pressure relief and the piping of relief back to primary containment are possible considerations. As identified in SECY-90-016, each of these high-to-low-pressure interfaces must also include the following protection measures: (1) the capability for leak testing of the PIVs, (2) assurance that the valve position operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure system and both isolation valves are not closed.

In CESSAR-DC Section 20.2.29, ABB-CE states that CESSAR-DC Appendix 5E addresses design improvements to minimize the possibility of a ISLOCA outside the containment. ISLOCA events are discussed in Sections 19.1 and 19.2 of this report.

Responding to staff RAI Q440.45, which is listed in Appendix B of this report, ABB-CE submitted its evaluation of various interfacing systems (i.e., CVCS, process sampling system (PSS), seal injection, seal bleedoff, SIS, and SCS). ABB-CE concluded that the design of systems and subsystems interfacing with the RCS will be in full compliance with the requirements specified in SECY-90-016, as discussed for ISLOCA protection. However, certain portions of these systems do not meet the RCS URS criteria and no technical basis was offered to justify why it is impractical to further reduce pressure challenge. Also, ABB-CE's discussion of the ISLOCA protection did not include the associated flanges, connectors, packings (including valve stem seals), pump seals, heat exchangers tube, valve bonnets, instrumentation lines, RCS drain, and vent. The staff required that these two areas be addressed in accordance with the ISLOCA requirements in SECY-90-016 discussed above. In response, ABB-CE submitted a report on June 15, 1992 (Report CE NPSD-741-P, "Evaluation of Design Features Which Minimize Probability of Interfacing LOCAs for System 80+ Standard Design") on design features that minimize the probability of an ISLOCA for the System 80+ design. Because the staff did not complete its evaluation of this ABB-CE report for the DSER (DSER Open Item 20.2-14), the staff stated in the DSER that it would provide its evaluation of CE NPSD-741-P in this report.

In Amendment Q of CESSAR-DC, ABB-CE submitted CESSAR-DC Appendix 5E, "Evaluation of the System 80+ Standard Design to Interfacing System LOCA Challenges," which superseded CENPSD-741-P. In Appendix 5E, ABB-CE evaluates the vulnerability of the System 80+ design to ISLOCAs. All low-pressure systems that are directly or indirectly connected with the RCS were examined, including the pressurization pathways that are established by an inadvertent opening of a valve or valves, a failure of containment isolation, or the postulation that valves are fully open. The specific components, such as flanges, valves, pump seals, heat exchangers, vents, and drains were also evaluated. Therefore, DSER Open Item 20.2-14 is resolved.

The systems directly connected to the RCS during some modes of operation are the SCS, SIS, CVCS, and PSS. Each of these systems and associated subsystems was evaluated for compliance with the ISLOCA criteria below.

As a result of the ISLOCA evaluation, ABB-CE made design improvements to the interfacing low-pressure systems. For those systems not meeting the ISLOCA criterion, system design modifications were made by either increasing the design pressure of the low-pressure systems or equipment to a high-pressure rating of 40 percent of RCS normal operating pressure, that is, 6.31 MPa (900 psig), or incorporating design features that terminate or limit the scope of ISLOCA events. Increasing the low-pressure system design pressure to the high-pressure rating ensures that the URS is at least equal to the RCS pressure, and therefore, complies with the staff position described in SECY-90-016. When the high-pressure rating is not designed for, ABB-CE provided justifications or bases for impracticality to design to such a pressure, and implemented system modifications, such as relocation of low-pressure interface, isolation of a low-pressure subsystem in the pressure pathway from the pressure source, and pressure relief to limit the pressurization within the design capabilities of the low-pressure subsystem.

System modifications have been made to relocate certain low-pressure interfaces with the SCS and the CSS. For example, the connection from the in-containment reactor water storage tank (IRWST) through the SCS to the CVCS makeup system is deleted and replaced with dedicated lines to allow the makeup system direct access to the IRWST without going through the SCS. This eliminates the potential for an ISLOCA to the low-pressure makeup system from the SCS, and separates the boron adjustments to the IRWST from the SCS and CSS operation. Also, the design pressure of all remaining sections outside the containment has been increased to 6.31 MPa (900 psig) for the SCS, CSS, and SIS. This high-pressure rating design also applies to the associated elements and components, such as gasketed flange connections valves, valve bonnet seals, pump seals, and heat exchangers tubes. Therefore, they are in compliance with the ISLOCA requirements of SECY-90-016. However, certain portions in the CVCS and PSS are still not designed for the high-pressure rating. ABB-CE provided justifications for not designing to the higher pressure rating and implemented design improvements to reduce pressure challenges. As discussed below, the staff has concluded that they are acceptable.

In the CVCS, the design pressures for the letdown and charging sides are 17.24 MPa (2485 psig) and 20.97 MPa (3025 psig), respectively. However, certain portions, that is, the low-pressure sections downstream of the flow control valve in the letdown line, the outermost section in the RC pump bleedoff line upstream of the volume control tank (VCT), the section of charging line upstream of the charging pumps, and the filter vent and drain lines in the RCP seal injection line upstream of the equipment drain tank (EDT), are not designed to the high-pressure rating of 6.31 MPa (900 psig). These low-pressure portions have design pressures comparable to the design pressure of connected low-pressure tanks and components, such as VCT, EDT, boric acid batching tank, holdup tank, reactor makeup water tank, and ion exchangers.

These low-pressure tanks and ion exchangers in the CVCS are used during various plant operation modes to process water, and have design pressures well below 6.31 MPa (900 psig). Increasing the design pressure of these tanks and components to 6.31 MPa (900 psig) could require an increase of wall thickness by a factor of two to ten. Several large tanks are field fabricated and would require new fabrication technology to accommodate the increased plate thickness and support structure. There are also many low-pressure systems interfacing with the CVCS, such as hydrogen supply system and radioactive waste management systems. Increasing the design pressure of the low-pressure portion of the CVCS does not terminate, mitigate, or even control the scope of an ISLOCA event, unless the interfacing systems are also designed to a higher pressure. ABB-CE asserts that increasing the design pressure of the entire CVCS to 6.31 MPa (900 psig) is impractical because of the increased complexity of overall system, fabrication technique, and the impact to the low-pressure systems interfacing with the CVCS. Therefore, alternative design improvements were made to reduce the ISLOCA challenges. The staff concludes that ABB-CE has properly justified not designing the entire CVCS to the high-pressure rating as long as proper design improvements are implemented to minimize the ISLOCA challenges.

ABB-CE has made the following design improvements to the CVCS: (1) increase the charging pump's suction piping (downstream of the check valves) design pressure to 6.31 MPa (900 psig), (2) reduce RCP seal injection filter vent and drain line sizes to limit flow to within makeup capacity, (3) add a pressure sensor-controller to the letdown line located downstream of the flow control valves, and (4) add a pressure sensor-controller to the charging line in the common suction line. These added pressure sensors-controllers give the operator information that an ISLOCA is occurring, and provide signals for automatic closure of the containment isolation valves (CIVs) to terminate the ISLOCA challenges by preventing any further pressure communication on both sides of the CIVs.

The impact of using the pressure controllers as an ISLOCA remedy was evaluated considering (1) spurious actuation of the pressure controller causing the CIVs to close, and (2) failure of the pressure controllers to isolate the low-pressure section of the CVCS. The effect of spurious isolation of the letdown or charging line CIVs, which has a low probability estimated by ABB-CE to be about 3×10^{-4}/year, was determined to be insignificant on CVCS operation because (1) letdown is not required for normal operation, (2) the plant can still maintain Mode 1 operation without charging flow or the seal injection flow if CCW is supplied for the RCPS. The consequence of failure to close the CIVs to isolate the low-pressure sections is considered acceptable because the relief valves in the low-pressure section of the letdown and charging lines serve as backup to these occurrences. ABB-CE states that the setpoint for the pressure controllers is selected to close the CIVs before the setpoints of the relief valves on the letdown and charging lines are reached. This prevents the relief valves from opening when the pressure controllers and CIVs operate properly, thereby avoiding an ISLOCA should the relief valves stay open.

ABB-CE asserts that these design improvements preserve system integrity because the low-pressure portions of the letdown and charging lines and all other interfacing low-pressure systems are not pressurized by an ISLOCA event, and, therefore, do not have their integrity challenged. In addition, both letdown and charging lines satisfy the requirements of SECY-90-016 by providing (1) a high-pressure alarm in the control room to warn the operators of the events, (2) containment isolation valve position indication and control in the main control room, and (3) periodic leak testing of the containment isolation valves.

The RCP bleedoff line is discharged to the VCT, which is protected from over-pressurization by a relief valve discharging to the EDT. There is also a fixed resistance flow control orifice upstream to limit pressure and limit the bleedoff flow to 14.76 L/minute (3.9 gpm), which is within the makeup capability and can be collected in the EDT for more than 500 minutes before requiring operator action. The seal injection filter vent and drain lines are connected to the equipment drain header and EDT. The design improvement to reduce the filter vent and drain line size to limit the discharge rate to a fraction of the makeup capability prevents a rapid fill of the EDT, and allows sufficient time for an operator to terminate the event. As the low-pressure sections of the RCP seal injection and bleed lines are designed with pressure relief to the EDT, they would not be subjected to high-pressure challenges. Furthermore, the liquid level, temperature, and pressure of the VCT and EDT indicate and alarm in the main control room.

The PSS has direct connections to the RCS hot leg, pressurizer surge line, and the pressurizer steam space. These connection lines have high-pressure section with a design pressure of 17.24 MPa (2485 psig), and low-pressure sections with a design pressure of 1.48 MPa (200 psig) upstream of the VCT. Each sampling line has a fixed-resistance, flow-restricting orifice and small line size upstream to limit the flow and pressure, and protect the low-pressure portion of the sampling system from overpressurization during operation. Because the process radiation monitor in the sampling system is designed to withstand pressure no greater than 1.83 MPa (250 psig), designing the entire sampling system to the higher pressure rating would require a radiation monitor design that is not commercially available. As an alternative, ABB-CE made a design improvement by adding a pressure-relief valve upstream of the flow indicator with relief flow directed to the EDT to protect the low-pressure portion for the postulated event in which the discharge valve to the VCT is closed. The relief valve is sized to pass a flow rate equivalent to the sample line flow rate. With this flow rate, the flow control orifice upstream of the pressure-relief valve will create a pressure drop that will limit the pressure to an acceptable value. Furthermore, in conjunction with the indication of containment isolation valve position, the EDT has indication and alarm for liquid level, temperature and pressure in the control room to alert the operator, with sufficient time for operator action to terminate the event. The staff finds this design improvement acceptable.

The staff concludes that the System 80+ design is consistent with the staff position discussed in SECY-90-016 regarding ISLOCA and, therefore, DSER Open Item 20.2-14 is resolved.

By staff RAI Q210.81, listed in Appendix B of this report, and DSER Open Item 20.2-15, the staff requested that ABB-CE commit to perform preservice and periodic inservice leak testing of all safety-related RCS PIVs to verify the leak tight integrity. ABB-CE responded by letter dated February 12, 1992, committing to comply with the testing requirements specified in RG 1.68 and the ASME Code, Section XI. In CESSAR-DC Section 3.9.6.2.4, ABB-CE lists RCS PIVs. ABB-CE also states that those PIVs will be leak-rate tested in accordance with CESSAR-DC Table 3.9-15 and the surveillance requirements specified in TS 3.4.13.1. The staff concludes that the RCS PIVs leak testing requirements for the System 80+ design is acceptable as discussed in Section 3.9.6.2.4 of this report. On this basis, Issue 105 is resolved for the System 80+ design and DSER Open Item 20.2-15 is resolved.

Therefore, Issue 105 is resolved for the System 80+ design.

Issue 106: Piping and Use of Highly Combustible Gases in Vital Areas

Issue 106, in NUREG-0933, addresses the issue of combustible gases accumulating in buildings containing safety-related equipment. Except for hydrogen, most combustible gases are used in limited quantities and for relatively short periods of time. Hydrogen is stored in high-pressure storage vessels and is supplied to various systems in the auxiliary systems building through small-diameter piping. A leak or break in this piping could result in a combustible or explosive mixture of air and hydrogen, posing a potential loss of safety-related equipment. Issue 106 has not been resolved and was assigned a medium priority in NUREG-0933.

As stated in CESSAR-DC Section 20.2.30, the System 80+ design incorporates various compressed-gas systems as described in CESSAR-DC Section 9.5.10. The compressed-gas systems provide a variety of gases (e.g., hydrogen and nitrogen) under pressure, for numerous plant operating applications, including welding equipment, instrumentation, system purging, inerting, and diluting.

The systems typically consist of high-pressure gas cylinders, pressure regulators, and piping to distribute the gases throughout the plant. These non-safety-related compressed gas systems are designed to ensure that their failure does not jeopardize the operation of any safety-related system or component or both (see CESSAR-DC Section 9.5.10.1). Furthermore, with respect to the compressed-hydrogen system, the system is designed to be isolable and a leak detection system is included.

ABB-CE states in CESSAR-DC Section 9.5.10 that the requirements of SRP Section 9.5.1, "Fire Protection Program," apply to the hydrogen lines located in safety-related areas. These lines are either designed to seismic Category I requirements, sleeved so that the outer piping is vented to the outside, or are designed with excess flow check valves so that, in case of a line break, the hydrogen concentration in the affected area will not exceed 2 percent. The update of CESSAR-DC to add the requirements of SRP Section 9.5.1 was identified as DSER Confirmatory Item 20.2-1. Therefore, DSER Confirmatory Item 20.2-1 is resolved.

In the DSER, the staff indicated that ABB-CE's response provided an acceptable interim resolution to the issue and indicated that the COL applicant would address the final resolution to this issue if determined. This is DSER COL Action Item 20.2-6. The staff has reviewed ABB-CE's revised response discussed above and finds that it is now acceptable. Therefore, COL Action Item 20.2-6 is resolved.

Based on the above, Issue 106 is resolved for the System 80+ design.

Issue 113: Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

Issue 113, in NUREG-0933, addressed the staff's concerns in 1985 that there were no requirements for dynamic qualification testing or surveillance testing of LBHSs (i.e., > 50 kips load rating). The safety concern was the integrity of the SG lower support structures when subjected to a seismic event; however, this issue was applicable to all components, structures, and supports that rely on this type of snubbers for restraint from seismic loads and other dynamic loads, such as high-energy line breaks and water hammers.

LBHSs are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events, yet also allow sufficient piping component flexibility to accommodate system expansion and contraction from such thermal transients as normal plant heatups and cooldowns. Dynamic testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. Issue 113 was resolved with no new requirements.

In the response dated January 29, 1992, to staff RAI Q210.89, which is listed in Appendix B to this report, concerning the dynamic qualification testing of LBHSs, ABB-CE stated that this issue is superseded by Issue A-13, "Snubber Operability Assurance." In its resolution to Issue A-13 in CESSAR-DC Section 20.2.55, ABB-CE states that it intends to minimize the use of snubbers using optimization procedures outlined in CESSAR-DC Section 3.9.3.4. ABB-CE outlined a program intended to ensure snubber operability that includes such elements as consideration of load cycles and total expected travel for the life of the snubber, visual inspection and measurement of thermal movements during startup tests, and a snubber ISI and testing program which includes periodic maintenance and visual inspection following a faulted event, a functional testing program with replacement or repair of snubbers failing inspection, or test criteria.

In the DSER, the staff did not accept that Issue 113 is superseded by Issue A-13. There was no statement in NUREG-0933 on Issue 113 that this is true. Issue 113 specifically addressed the dynamic qualification testing of LBHSs as opposed to the more general snubber design and operability criteria proposed by ABB-CE for resolution of Issue A-13. This was identified as Open Item 20.2-16. See the discussion on Issue A-13 in Section 20.2 of this chapter.

ABB-CE, in Amendment L to the CESSAR-DC, has proposed dynamic qualification testing for LBHSs up to test system capability and, for snubbers exceeding test facility limitations, have their characteristics calculated based upon dynamic test data of a similar snubber qualified by testing. Therefore, DSER Open Item 20.2-16 is resolved and Issue 113 is resolved for the System 80+ design.

Issue 118: Tendon Anchorage Failure

Issue 118, in NUREG-0933, addressed the concerns raised by the staff after inspections at Farley Unit 2 in 1985, about three lower vertical tendon anchor heads for the concrete containment structure that were found broken. A tendon inspection and surveillance program was initiated at both Farley units, and the licensee evaluated the cracked tendon anchor heads and concluded that the containment had never been lost its structural integrity. However, the failure of anchor heads to carry the tendon forces could have jeopardized the structural integrity of the containment during an accident. RGs 1.35 (Rev. 3) and 1.35.1 resolved this issue.

In CESSAR-DC Sections 3.8, 6.2, and 20.2.32, ABB-CE states that the containment is a steel structure. RGs 1.35 and 1.35.1 are for concrete containment structures; therefore, they do not apply to the System 80+ design. Based on this, Issue 118 is resolved for the System 80+ design because the containment is a steel structure.

Issue 119.1: Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads

Issue 119, "Piping Review Committee Recommendations," in NUREG-933, was the comprehensive review of NRC requirements, requested by the NRC Executive Director for Operations (EDO) in 1983, in the area of plant piping in safety-related systems and high energy lines important to safety. The NRC Piping Review Committee (PRC) reviewed and evaluated then-existing regulatory requirements to provide recommendations on additional requirements and identify areas requiring further action. Issue 119.1 comprised the following three PRC Category A recommendations in NUREG-1061 ("Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volumes 2 and 3, dated April 1985 and November 1984, respectively): (1) LBB (A-1), (2) decoupling of seismic and LOCA Loads (A-5), and (3) complete research on decoupling (A-4).

Decoupling of SSE and LOCA loads has never been adopted by the NRC, and combination of these two loads is required as indicated in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures." However, the elimination of LOCA dynamic loads from this load combination may be achieved through application of LBB. This approach is acceptable if certain limitations and requirements as stated in GDC 4 are met and if acceptance criteria contained in NUREG-1061 (Volume 3) are satisfied.

In CESSAR-DC Section 3.9.3.1 and Tables 3.9-10 through 3.9-14, SSE and pipe break loads are combined, which is acceptable as indicated in Section 3.9.3 of this report. In CESSAR-DC Section 3.6.2.1.3, the methodology for application of LBB to the main coolant loop, surge line, main steamline, SI line, and shutdown cooling line is described. In addition, bounding analyses for establishing limits of LBB application for these lines are completed during the System 80+ design certification phase and will be verified during the combined license phase (Subpart C to 10 CFR Part 52) by performing the appropriate ITAAC.

The staff evaluation of LBB application to the System 80+ design is presented in Sections 3.6.3 and 3.12.8 of this report. The staff finds that ABB-CE's position on decoupling of seismic and LOCA loads is acceptable. Therefore, on this basis, Issue 119.1 is resolved for the System 80+ design.

Issue 119.2: Piping Damping Values

Issue 119.2, in NUREG-0933, addressed the recommendation of the PRC on how the damping values used in seismic dynamic analysis of nuclear power plant piping systems should be modified so that the piping reliability and plant safety can be enhanced by reduction of snubbers, less restraint to thermal expansion, and less obstruction to ISI of piping welds. In NUREG-1061, Volume 2, dated April 1985, the PRC recommended an interim position of using the PVRC damping (or the ASME Code Case N-411 damping) as an alternative to the damping values in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

In RG 1.84, "Design and Fabrication Code Case Acceptability ASME Section III, Division 1," NRC endorsed the use of N-411 damping as an alternative to RG 1.61 damping values with certain limiting conditions.

In CESSAR-DC Section 3.7.1.3 and Appendix 3.9A, damping values to be used in piping dynamic analysis are presented. The values are in accordance with RG 1.61. In addition, ABB-CE states that damping values of Code Case N-411 may be used as an alternative when the response spectrum method of analysis is used and will be subject to conditions given in RG 1.84.

The staff evaluation of System 80+ piping damping values is in Sections 3.7.2 and 3.12.5.4 of this report. On the basis of this evaluation, the staff finds that the ABB-CE position on damping values is acceptable and Issue 119.2 is resolved for the System 80+ design.

Issue 119.3: Decoupling the Operating Basis Earthquake from the SSE

Issue 119.3, in NUREG-0933, addressed the concern of assuring public safety when the magnitude of the operating basis earthquake (OBE) is established at a level different from that specified in Section V(a)(2) in Appendix A to 10 CFR Part 100. The regulation establishes the maximum vibratory ground acceleration of the OBE as at least one-half of the maximum vibratory ground acceleration of the SSE.

In SECY-90-016, the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the SSE. The Commission approved the staff's position in its SRM dated June 26, 1990. Subsequently, the staff requested in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 3, 1993, that the Commission approve altogether the elimination of the OBE from the design of SSCs in ALWRs. The Commission approved the request in its SRM dated July 21, 1993. The revision to 10 CFR Part 100 (Appendix A) would allow, as an option, that the OBE be eliminated from design when the OBE is established at less than or equal to one-third the SSE. Therefore, the need to decouple the OBE from the SSE is no longer an issue for the System 80+ design because the OBE has been eliminated for the System 80+ design. The staff's evaluation of the elimination of the OBE from the System 80+ design is addressed in Section 3.1 of this report.

Based on the above, Issue 119.3 is resolved for the System 80+ design.

Issue 119.5: Leak Detection Requirements

Issue 119.5, in NUREG-0933, addresses the NRC PRC regulatory recommendation A-6 in NUREG-1061 (Volume 1, dated August 1984) to improve leak detection systems for the RCPB and the effects of the adoption of LBB by NRC for primary piping on leak detection system design requirements. A review of leak detection systems and LERs on these systems at then-current operating plants was reported in NUREG/CR-4813 ("Assessment of Leak Detection Systems for LWRs," January 1987, Revision 1, October 1988). The staff concluded that existing RCPB leak detection systems conforming to the guidance of RG 1.45, "Reactor Coolant Pressure Boundary Leakage

Detection Systems," and SRP Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," are adequate for the purposes of LBB in the great majority of situations. The principal deficiency of these systems is that they provided no information on the location of a leak, which had to be found visually after plant shutdown.

Another part of the NRC effort to develop a resolution to this issue was work performed by Argonne National Laboratory to develop an advanced ALMS. This work, reported in NUREG/CR-5134 ("Application of Acoustic Leak Detection Technology for the Detection and Location of Leaks in Light Water Reactors," October 1988), indicated that such a system appears capable of locating as well as quantifying pipe leaks.

As stated in NUREG-0933, the NRC has not developed a resolution to this issue and is considering revisions of RG 1.45 and SRP Section 5.2.5 to apply the results of the above investigations to regulatory requirements. Nevertheless, the NRC staff has determined that an acceptable resolution of Issue 119.5 for the System 80+ design is compliance with SRP Section 5.2.5 and the guidance of RG 1.45. ABB-CE states in CESSAR-DC Section 20.2.36 that, for the resolution of this issue, the RCPB leak detection systems shall be consistent with SRP Section 5.2.5 and RG 1.45.

Specifically, identified leakage (i.e., from sources that cannot practically be made 100-percent leaktight, such as valve stem packing glands) shall be collected and monitored separately from unidentified leakage. Unidentified leakage shall be collected and monitored by at least three out of four independent methods described in RG 1.45. The methods should have a sensitivity adequate to detect a leak of 3.8 L/minute (1 gpm) in less than 1 hour. Indicators and alarms for each leakage detection system shall be provided in the control room.

The RCPB leakage detection systems of the System 80+ design are described in CESSAR-DC Section 5.2.5, and are consistent with the recommendations of RG 1.45. Correlation of the above leakage detection capability for "unidentified" leakage with LBB analyses for evaluating pipe crack stability is discussed in CESSAR-DC Section 3.6.3. Collection and measurement of "identified" leakage is described in CESSAR-DC Section 5.2.5.1.2. The staff concluded that CESSAR-DC Sections 3.6.3 and 5.2.5 are acceptable, as documented in Sections 3.6.3 and 5.2.5 of this report.

Four independent methods of detecting "unidentified" leakage, including three of those recommended in RG 1.45, are provided as follows, and are described in CESSAR-DC Section 5.2.5.1.1:

(1) The reactor coolant inventory method is used to detect large volume leakage over a period of steady-state operation by continuously monitoring the net makeup flow to the RCS. Since letdown flow and the RCP seal bleedoff flow are collected by the CVCS and recycled into the RCS, net makeup flow is trended in the control room by the Nuplex 80+ DPS.

(2) The primary method designed to detect leakage rates as low as 3.8 L/minute (1 gpm) in less than 1 hour is by monitoring the rates of change of the sump water levels in the containment holdup volume and the reactor vessel cavity, together with the discharge rates and running times of the sump pumps. The Nuplex 80+ DPS integrates these measurements and calculates the leak rate. Control room alarms are activated if a leak rate greater that 3.8 L/minute (1 gpm) is calculated and also if a reactor cavity sump pump starts, since under normal conditions no leakage is expected into the reactor cavity.

(3) A containment gaseous radiation monitor measures the gamma radioactivity levels in the containment atmosphere by continuous sampling. Leakage is detected by this method and, to the extent practicable, quantified, with a response time dependent on such factors as the fraction of failed fuel, the fission product inventory in the core, and time of transit from the origin of the leak to the monitor. The activity is indicated in the control room by the DPS and averaged hourly.

(4) A containment air particulate monitor measures the containment atmosphere particulate beta-radioactivity by continuous sampling. The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent on the magnitude of the normal baseline leakage into the containment, and the reactor coolant activity. The particulate activity concentration is indicated in the control room by the DPS, and averaged hourly. High activity activates an alarm. The airborne particulate monitoring is designed to remain functional during and after a SSE, as recommended in RG 1.45.

In addition to the methods described above, the System 80+ design has an ALMS designed to meet, in part, the guidance of RG 1.45. The function of the ALMS is to detect a leak at specific locations or within specific components of the RCS. The ALMS is described in CESSAR-DC Section 7.7.1.6. As described in Section 5.2.5 of this report, the NRC staff has determined that the System 80+ design is consistent with the guidance of RG 1.45.

Therefore, Issue 119.5 is resolved for the System 80+ design.

Issue 120: On-Line Testability of Protection Systems

Issue 120, in NUREG-0933, addressed requirements for at-power testing of safety system components without impairing plant operation. These requirements apply to both the RPS and ESFAS. A protection system with two-out-of-four (2/4) logic that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three (2/3) logic configuration meets this requirement.

ABB-CE states in CESSAR-DC Section 20.2.37 that, for the resolution of this issue, the System 80+ design has all-digital I&C systems, described in Chapter 7 of CESSAR-DC, that allow on-line testing of the systems. The System 80+ RPS and ESFAS, which are evaluated and approved in Sections 7.4 and 7.3 of this report, respectively, are 2/4 logic systems that allow one channel to be placed in bypass for testing and maintenance, while the other three channels operate as a 2/3 logic system.

Therefore, Issue 120 is resolved for the System 80+ design.

Issue 121: Hydrogen Control for Large, Dry PWR Containments

Issue 121, in NUREG-0933, documented the staff's research on hydrogen control in large, dry PWR containments. In response to the TMI-2 accident, the Commission issued regulatory requirements on hydrogen control in 10 CFR 50.34 and 50.44. A hydrogen control system is required by 10 CFR 50.34(f) based on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment, or a postaccident atmosphere that will not support hydrogen combustion; however, only those plants whose CPs had not been issued at the time of the TMI-2 accident were covered by 50.34(f).

In SECY-90-016, the staff recommended to the Commission that the hydrogen control requirements for evolutionary plants be identical to those stated in 10 CFR 50.34(f). This regulation specifically requires a hydrogen control system that can safely accommodate an amount of hydrogen equivalent to that generated by the reaction of 100 percent of the fuel-clad metal and that can ensure that uniformly distributed hydrogen gas concentrations in the containment do not exceed 10 percent by volume. The Commission approved the staff's recommendation in a Commission SRM dated June 26, 1990. In its letter dated December 6, 1991, EPRI stated that it would modify the URD for evolutionary plants to comply with the staff position of 100-percent active fuel cladding and a maximum concentration of 10 percent of hydrogen in the containment.

ABB-CE states in CESSAR-DC Section 20.2.38 that, for the resolution of this issue, the System 80+ design includes a hydrogen mitigation system (HMS) for control of combustible gas concentration in the containment during and following a degraded-core accident. The HMS, which is designed in accordance with the requirements of 10 CFR 50.34(f), is described in CESSAR-DC Sections 6.2.5 and 19.11.3.4, and Appendix 19.11K. The HMS was evaluated in Sections 6.2.5 and 19.2.3.3.1 of this report.

The HMS consists of a system of igniters installed in the containment to allow adiabatic, controlled burning of hydrogen at low concentration to preclude buildup to detonable concentration levels. Using a global distribution of igniters, the system is expected to prevent the average hydrogen concentration from reaching 10 percent by volume during a degraded-core accident with 100-percent fuel-clad metal-water reaction. The igniters are ac-powered glow plugs and are divided into two redundant groups, each group having independent and separate circuits and circuit breakers. The igniters in each group are located so as to ensure adequate coverage in the event of a single failure. If there is a LOOP, the igniters can be powered from the AAC source (combustion turbine) or the EDGs. They can also be powered from the Class 1E emergency batteries through dc-to-ac inverters. The igniters are manually activated from the control room.

The spherical steel containment (see CESSAR-DC Section 3.8), which has a diameter of 61 m (200 ft) and a free volume of approximately 96 x 10^3 m^3 (3.4 x 10^6 ft^3), and its internal structures are designed to promote mixing by natural circulation and to minimize localized concentrations of hydrogen. The HMS igniters are positioned near areas in which hydrogen may accumulate most rapidly.

HMS components in the containment are capable of sustaining normal operational and seismic loads. The HMS is not required to function in a DBA and is not safety grade. However, the system is expected to mitigate the effects of a degraded-core accident and is designed to withstand the appropriate environmental conditions. Equipment essential to mitigate, manage, and monitor the accident and shut down the plant is identified. A best-estimate determination of the environment (including the effects of HMS activation) to which this equipment will be exposed during the accident is then made. Survivability of the essential equipment is evaluated based on direct comparisons with existing qualification data or through experience with similar types of equipment. This process is evaluated by the staff in Section 19.2.3.3.6 of this report.

Preoperational testing and periodic operational testing of the HMS discussed by ABB-CE in CESSAR-DC Section 6.2.5 and Chapter 16 on plant TSs ensure the operability of the system.

COL Action Item 20.2-7 in the DSER identified the requirement for the staff to review relevant plant-specific design features regarding combustible gas control for conformance to 10 CFR 50.34(f) when an COL application is received. The staff's review of this issue finds that the System 80+ design meets the requirements of SECY-90-016 and 10 CFR 50.34(f) for hydrogen control (see Section 19.2.3.3.1 of this report). Therefore, COL Action Item 20.2-7 is resolved.

Therefore, Issue 121 is resolved for the System 80+ design.

Issue 122.2: Initiating Feed and Bleed

Issue 122, in NUREG-0933, investigated the findings of the NRC inspection in 1985 of the loss-of-feedwater event at Davis Besse on June 9, 1985. The Issue 122.2 dealt with the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following the loss of the SG heat sink (i.e., loss of feedwater). In an analysis of the loss-of-feedwater event, the staff found that operators were hesitant to initiate feed-and-bleed operations, and that the control room instrumentation was inadequate to alert operators to the need to initiate feed and bleed. A loss of feedwater in combination with a failure to diagnose and take corrective actions (i.e., initiate feed and bleed) would result in a loss of core cooling. The staff raised Issue 122.2 and requested that applicants of an advanced reactor design provide instrumentation of sufficient reliability to correctly identify a total loss of feedwater and mitigate its consequence using such strategies as feed and bleed.

ABB-CE addresses this issue in CESSAR-DC Section 20.2.39. The System 80+ design includes the postaccident monitoring instrumentation (PAMI) for identifying and mitigating accidents. The PAMI is itemized in CESSAR-DC Section 7.5.1.1.5 and Table 7.5-3, and includes the parameters monitored, the number of sensed channels, sensor ranges, and location and equipment qualification requirements. The plant parameters monitored to identify a total loss of feedwater are main and EFW flow, reactor coolant temperature, pressure and degree of subcooling, and SG pressure and level (wide range).

The SDS design, as described in CESSAR-DC Section 6.7, supplies the feed-and-bleed function for beyond-design-basis events.

To address Issue 122.2, ABB-CE referred to the resolutions for Issue I.C.1, in Section 20.3 of this report; regarding criteria for feed-and-bleed initiation. In review of these resolutions, the staff found that the current ABB-CE EOGs included in CEN-152, "Combustion Engineering Emergency Procedure Guidelines," Revision 3, dated May 1987, gave adequate initiation criteria for feed and bleed in appropriate recovery procedure guidelines (see Section 6.7.1 of this report for the resolution of DSER Open Item 6.7.1-2), and the information in CEN-152 is sufficient and clear for the owner/operator to prepare the plant-specific operating procedures by using feed and bleed to mitigate an accident. Procedures are discussed in Sections 13.5 and 18.7 of this report.

Also, in response to staff RAI Q440.23 (listed in Appendix B of this report) regarding acceptability of CEN-152 to the System 80+ design, ABB-CE committed to add the design enhancements, including the SDS in the updated EOGs. The resolutions of ABB-CE are acceptable because (1) ABB-CE provided adequate guidelines for mitigation of the feed-and-bleed operation in its current EOGs, (2) ABB-CE updated EOGs to include the SDS design for the System 80+ plant, and (3) the review of the updated EOGs is covered by Issue I.C.1 in Section 20.4 of this report.

Therefore, Issue 122.2 is resolved for the System 80+ design.

Issue 124: Auxiliary Feedwater System Reliability

Following the loss of feedwater event at Davis Besse in 1985, Issue 124, in NUREG-0933, addressed increasing reliability of the auxiliary or EFW system to 10^{-4} unavailability/demand. In 1985, operating experience as well as staff and industry studies indicated that these systems failed at a high rate. A function of this system in the majority of current plants is to supply water to the secondary side of the SGs during system fill, normal plant heatup, normal plant hot standby, and normal plant cold shutdown. The EFW system also functions following loss of normal feedwater flow, including loss due to offsite power failure, and supplies EFW following such postulated accidents as a MFW line break or a MSLB. Therefore, the reliability of this system is important to plant safety.

The NRC investigation of the Davis Besse event indicated that the potential inability to remove decay heat from the reactor core was due to the questionable reliability of the EFWS caused by any or all of the following:

- loss of all EFW due to common-mode failure of the pump discharge isolation valves to open
- excessive delay in recovering EFW because of a difficulty in restarting the pump steam-driven turbines once they tripped
- interruption of EFW flow because of failures in steamline break and feedline break accident mitigation features

In addition, the investigation of the event indicated that (1) a two-train system with a steam turbine-driven EFW pump may not be able to achieve the desired level of reliability and (2) the provision to automatically isolate EFW from a SG affected by a main steamline or feedwater-line break may tend to increase the risk that adequate DHR is not available, rather than to decrease it.

ABB-CE states in CESSAR-DC Section 20.2.40 that the System 80+ EFW system is designed to maintain a high level of availability and reliability consistent with its importance as a safety system. The reliability and design features are described in CESSAR-DC Section 10.4.9, and include two independent trains with each train aligned to supply its respective SG. Each train consists of

- one emergency feedwater storage tank (EFWST)
- one 100-percent-capacity motor-driven pump subtrain and one 100-percent-capacity steam-driven pump
- flow control valve
- isolation valve
- check valve
- a cavitating venturi
- specified instrumentation

One design feature of the EFWS system that improves its reliability is its component and piping separation and diversity. For example, each subtrain is separated from the other and, therefore, has its own discharge line through the SG isolation valve and check valve. In addition, the pump crossover lines contain redundant, locked-closed, isolation valves. The subtrain design reduces the potential for single failure and improves system reliability.

Because of the improved reliability of the EFWS design, the unavailability for the system was estimated from PRA studies to be in the range of 1×10^{-4} to 1×10^{-5} per demand, as described in CESSAR-DC Section 10.4.9.1.2. Analysis identified in CESSAR-DC Sections 10.4.9.1.2 and 19.6.3.7, which was developed using generic data, assesses the system's ability to function on demand and demonstrates its compliance with the unavailability range given above. Therefore, the EFWS meets the recommended unavailability goal of 1×10^{-4} per demand identified in SRP Section 10.4.9 (Rev. 2), "Auxiliary Feedwater System (PWR)."

In the DSER, the staff stated that the resolution of this issue was acceptable pending final resolution of the open and confirmatory items in DSER Section 10.4.9. As discussed in Section 10.4.9 of this report, the open and confirmatory items have been resolved.

Therefore, Issue 124 is resolved for the System 80+ design.

Issue 125.I.3: Safety Parameter Display System Availability

Issue 125, in NUREG-0933, addressed the long-term actions from NUREG-1154 , "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," dated July 1985, and the EDO memorandum dated August 5, 1985, on the loss-of-feedwater event at Davis Besse on June 9, 1985. Issue 125.I.03 addressed whether NRC requirements should be revised regarding the safety parameter display system (SPDS) availability and the reliability of the information it displays. The TMI-2 accident demonstrated the need for improving how information is relayed to the control room operators. As a result of Issue I.D.2 of the TMI Action Plan, which is discussed in Section 20.4 of this chapter, installation of a SPDS was required to provide control room operators continuous information from which the plant safety status could be readily and reliably assessed.

ABB-CE addressed the resolution of Issue 125.I.3 in CESSAR-DC Section 20.2.41. See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is acceptably resolved. In the DSER, the staff stated that this issue would be discussed in this report and designated the action as DSER Open Item 20.2-17. On the basis of this evaluation, DSER Open Item 20.2-17 is resolved.

Based on Section 18.3.3.2.5 of this report, Issue 125.I.3 is resolved for the System 80+ design. In Section 20.4 of this report, the staff concluded that Issue I.D.2 was resolved for the System 80+ design.

Issue 125.II.7: Reevaluate Provisions To Automatically Isolate Feedwater From Steam Generator During a Line Break

Issue 125, in NUREG-0933, addressed the long-term actions from NUREG-1154 and the EDO memorandum dated August 5, 1985, on the loss-of-feedwater event at Davis Besse on June 9, 1985. Issue 125.II.7 addressed the need for licensees to reassess the benefits of automatically isolating the EFW system after a break in the secondary side of the SG. For a typical PWR with automatic isolation (AI) of the EFW (AI-EFW), a low-SG-pressure signal causes closure of main steam isolation valves (MSIVs) and isolation of EFW from the faulted SG during a steamline break. AI-EFW minimizes blowdown from the SG secondary-side line break and limits primary system overcooling and the potential for return to criticality owing to positive moderator reactivity feedwater caused by overcooling of RCS inventory. If the EFW were not isolated, the peak containment pressure for secondary-side breaks would exceed that caused by a large-break LOCA, the DBE for the containment design.

However, AI-EFW has disadvantages. If both channels of the controlling isolation logic system were to spontaneously actuate, the availability of EFW would be lost and the MSIVs would close. For the plants using turbine-driven MFW pumps, the MFW pumps would be lost following the closure of the MSIVs and the loss of steam, and this loss would result in the loss of the secondary side heat sink. The capability to lock out the isolation logic is necessary to preclude such an event.

The staff determined (as stated in NUREG-0933) that, for a new plant, the design does not need to include automatic isolation of EFW following a steamline break or feedwater-line break, provided that the results of the analyses of the secondary-side line break and the containment analysis meet the applicable design criteria in the SRP, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3, dated July 1981.

ABB-CE describes the resolution of Issue 125.II.7 in CESSAR-DC Section 20.2.42. The System 80+ design does not include automatic SG isolation logic. The design has an EFWS, which provides an independent safety-related means of supplying quality feedwater to the SGs for removal of heat and prevention of reactor core uncovery during emergency phases of plant operation. EFW will be provided to both SGs during a depressurization event. The EFWS is a dedicated safety-related system which has no functions for normal plant operation (see CESSAR-DC Section 10.4.9).

The EFWS is designed to be automatically or manually initiated, supplying feedwater to the SGs for any event that results in the loss of normal feedwater and requires heat removal through the SGs, including the loss of normal onsite and normal offsite ac power. Four-channel control logic is provided, so that a single failure neither spuriously actuates nor prevents EFW supply. In addition, manually reset variable setpoints are used, to enable cooldown without actuating the MSIS.

The analyses to support the adequacy of the EFW design are discussed in CESSAR-DC Section 6.2 for the containment analysis and in CESSAR-DC Sections 15.1 through 15.6 for the transient analyses. The EFWS includes a design requirement that the EFW flow to each SG be restricted by a cavitating venturi to protect the EFW pump from damage from excessive runout flow. The EFW storage has a capacity of 1.32 x 10^6 L (350,000 gallons) from each of the two safety-related EFW storage tanks to achieve safe cold shutdown.

In the analysis, the assumption of the operator action delay time was consistent with the SRP, requiring that the operators not act to terminate the EFW flow to the faulted SG within 30 minutes of the break in the SG secondary system. The staff reviewed these analyses and concluded in Section 6.2 and Chapter 15 of this report that the analyses correctly reflect the design of EFW without the feature of automatic isolation logic, and that they demonstrate the compliance of the acceptance criteria specified in the related SRP sections regarding primary system overcooling, SG overfill, and containment overpressurization.

Operator action is also discussed in Chapters 18 and 19 of this report and in Issue B-17 of Section 20.2 of this chapter.

In the DSER, the staff concluded that ABB-CE's resolution of Issue 125.II.7 would be acceptable pending the resolution of open and confirmatory items identified in Section 10.4.9 of the DSER. All of the open and confirmatory items have been resolved, as discussed in Section 10.4.9 of this report.

Therefore, Issue 125.II.7 is resolved for the System 80+ design.

Issue 128: Electric Power Reliability

Issue 128, in NUREG-0933, addressed the following three issues that were related to the reliability of onsite Class 1E ac and dc power system: (1) Issue 48, "LCO [Limiting Condition for Operation] for Class 1E Vital Instrument Buses in Operating Reactors"; (2) Issue 49, "Interlocks and LCOs for Class 1E Tie Breakers"; (3) Issue A-30, "Adequacy of Safety Related DC Power Supplies." To provide a more integrated approach to resolving these three issues. The staff combined the three issues into Issue 128.

As stated above and by ABB-CE in CESSAR-DC Section 20.2.43, the resolution of Issue 128 is addressed in the resolutions of Issues 48, 49, and A-30 in this section and in Section 20.2 in this chapter. Because these three issues are concluded to be resolved for the System 80+ design, Issue 128 is resolved for the System 80+ design.

Issue 130: Essential Service Water Pump Failures at Multiplant Sites

Issue 130, in NUREG-0933, addressed the vulnerability of Byron Unit 1 to core-melt sequences in the absence of the availability of Unit 2 (not yet operational). While Unit 2 was under construction, it was necessary to make a third service water pump available to Unit 1 via a cross-tie with one of the two Unit 2 ESW pumps. This issue raised concerns relative to multiplant units that have only two ESW pumps per plant but have cross-tie capabilities. A limited survey of Westinghouse plants helped to identify the generic applicability of vulnerabilities of multiplant configurations with only two ESW pumps per plant. In the multiplant configurations identified (approximately 16 plants), all plants can share ESW pumps via a cross-tie between plants. Efforts to resolve included (1) a survey of Babcock and Wilcox and ABB-CE plants to determine if similar multiplant configurations with two ESW pumps per plant and cross-tie capabilities existed in these vendors' designs and (2) a survey of single-unit plants to determine if similar ESW vulnerabilities existed.

ABB-CE addresses this issue in CESSAR-DC Section 20.2.44. The System 80+ design is a single, independent plant design; that is, all systems and components necessary for the operation of the plant are dedicated to that particular plant. Therefore, the SSWS is designed for a single unit and does not rely on other systems or components from any other unit. In addition, the SSWS is an open-cycle system consisting of two redundant trains (four SSWS pumps) and is, therefore, reliable (see CESSAR-DC Section 9.2.1). The SSWS has the capability to dissipate the heat loads necessary for a safe reactor shutdown by rejecting heat delivered from the safety-related CCWS. The CCWS cools safety-related components, including those required for shutting the reactor down safely.

Where construction of multiple plants is desirable, separation and independence of all systems and components, including the SSWS, are maintained by the COL applicant and the architect-engineer (see CESSAR-DC Section 1.2.1.3).

In the DSER, the staff stated that it had not developed a final resolution on this issue. The NRC staff now concludes that the possibility of potential core damage from an SSWS failure as a result of shared systems and components is precluded because of the required separation and independence in the System 80+ design and in the SSWS design. In the DSER, the requirement for a COL applicant to address the final resolution of this issue, if determined, was identified as DSER COL Action Item 20.2-8. However, upon further review, the staff concludes that ABB-CE's response in CESSAR-DC Section 20.2.44 on this issue is acceptable and DSER COL Action Item 20.2-8 is not required. Thus, DSER COL Action Item 20.2-8 is resolved.

Therefore, Issue 130 are resolved for the System 80+ design.

Issue 135: Steam Generator and Steamline Overfill

Issue 135, in NUREG-0933, was initiated in 1986 to integrate various SG programs and related issues, including water hammer and steamline overfill from a SGTR. Because the staff concluded that SGTR and steamline overfill events are relatively low risks, this issue was resolved and no new requirements were established. The is documented in NUREG-0933 and NUREG/CR-4893, "Technical Findings Report for Generic Issue 135: Steam Generator and Steamline Overfill Issues," dated May 1991.

ABB-CE indicates in CESSAR-DC Section 20.2.45 that the design of the SGs will facilitate eddy current ISI of the tube bundle. The analyses of SGTR events are consistent with the guidance of SRP Section 15.6.3 as discussed in Section 15.3.9 of this report. ABB-CE also indicates that other related USIs and GSIs have been addressed. The plant is designed to minimize the probability of overfilling the SGs and main steam system during an SGTR event, and to mitigate the consequences of overfill should it occur.

A subissue in Issue 135 is the improved eddy current testing of SG tubes. The staff deferred this subissue to the development of a revision to RG 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator

Tubes." DSER Open Item 20.2-18 in the DSER is considered resolved as the COL applicant will be required by the System 80+ TS (Surveillance Requirement) 3.4.4.2 to develop an ISI program for SG tubes that will be based on eddy current testing techniques that have been demonstrated to have adequate detection and sizing capabilities.

The discussion above adequately addresses the SG overfill issue for the System 80+ design and, thus, Issue 135 is resolved for the System 80+ design.

See Issues A-4 and 66 for additional evaluations of SG issues on the System 80+ design.

Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits

Issue 142, in NUREG-0933, addressed observations in 1987 during SPDS evaluation tests that, for electrical transients below maximum credible levels, a relatively high level of noise could pass through types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy transmitted through the isolator could damage or seriously degrade the performance of the Class 1E components; in other cases, the electrically generated noise on the circuit may cause the isolation device to give a false output.

In resolving this issue, the staff determined from operating experience that isolation devices perform satisfactorily in the operating environment and have not been exposed to failure mechanisms that resulted in signal leakage. This was based, however, on plants that predominantly use electromechanical controls and may not be applicable to I&C systems with digital or electronic components. Therefore, it was recommended that an SRP section be written to provide review guidance for future plants that use digital systems.

The System 80+ design is a plant that uses digital systems. In CESSAR-DC Sections 7.1.1.7 and 20.2.46 for this issue, ABB-CE states that the RPS and ESFs component control system use fiber-optic technology for isolation between protection system channels, and equipment, cabinets, and operator interface devices in the main control room. Therefore, electrical isolators are not used in the System 80+ design; however, this issue raises the question of leakage through any isolators used in instrumentation circuits.

As discussed in Sections 7.1 and 7.2 of this report, the staff developed the acceptance criteria for digital systems using applicable international and national standards. The staff used the two-part approach given in SECY-92-053 ("Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992) to reach its safety finding for System 80+ design certification. In reviewing the I&C systems, the staff performed a detailed functional review of block diagrams of the I&C architecture to ensure the implementation of Commission requirements on digital systems including signal isolation. This review confirmed that the detailed functional requirements for the I&C systems were met. In Sections 7.2, 7.3, and 7.4 of this report, the digital RPS, engineered safety actuation system, and systems required for safe shutdown, respectively, were evaluated and the staff concluded that these systems were acceptable including signal isolation.

In Sections 7.7.1.4.1 and 7.7.1.7.2 of CESSAR-DC, on the discrete indication and alarm system and the DPS, ABB-CE addes that if a communication error occurs, an appropriate error message is generated and diagnostic tests are then applied to isolate the cause of the error. This would include errors caused by the leakage through a fiber-optic isolator.

Therefore, Issue 142 is resolved for the System 80+ design.

Issue 143: Availability of Chilled-Water Systems and Room Cooling

Issue 143, in NUREG-0933, addresses problems experienced in recent years at several nuclear plants, with safety system components and control systems that have resulted from a partial or total loss of HVAC systems. Many of these problems exist for two reasons: (1) the desire to provide increased fire protection and (2) the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been accomplished by enclosing the affected equipment in small, isolated rooms. However, the result has been a significant increase in the impact of the loss of room cooling. Another problem resulting from loss of room cooling is the advancement in control circuit design. With the introduction of electronic integrated circuits, plant control and safety have improved; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as RHR pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once these failures occur, the Advisory Committee on Reactor Safeguards believed that the impact of these failures on the proper functioning of air cooling systems has not been reflected in the final PRAs of plants. Thus, plants with similar, inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled-water systems (CWSs) to remove heat from the rooms containing the components. If chilled-water and HVAC systems are unavailable to remove heat, the ability of the safety equipment within the rooms to operate as intended cannot be assured.

Issue 143 has not been generically resolved and is classified in NUREG-0933 as a high-safety priority.

A possible solution to this issue would require a reevaluation of each plant's room heat load and heatup rate in order to locate areas in which the dependence of equipment operability on HVAC and room cooling may be reduced. Although the total elimination of this dependence may not be possible at all plants, this analysis would locate areas in which this dependence is critical. The critical dependencies and the ability to reduce them could be determined through the use of a plant-specific PRA. After the critical dependencies are identified, each plant would implement procedural changes (to provide alternate cooling) to eliminate or reduce the dependencies where possible. Hardware modifications may be needed for situations in which a procedure change cannot be implemented to reduce a critical dependency.

ABB-CE outlines the resolution of Issue 143 in CESSAR-DC Section 20.2.47 and states that System 80+ was designed to separate safety and non-safety equipment into separate systems. CWSs are described in CESSAR-DC Section 9.2.9, and HVAC systems are described in CESSAR-DC Section 9.4. Operability of these systems is ensured via the TSs.

The staff has reviewed the design of the System 80+ CWSs in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," as discussed in Section 9.2.9 of this report.

The CWSs are designed to provide and distribute a sufficient quantity of chilled water to air handling units (AHUs) in specific areas. The CWS is divided into the following two subsystems: an essential CWS (ECWS) that serves safety-related HVAC cooling loads, and a normal CWS (NCWS) that serves non-safety-related HVAC cooling loads. Additionally, in each ECWS division there is a ECWS heat exchanger and heat exchanger pump that allows the NCWS to supply 100 percent of the normal ECWS loads without directly connecting the water pathways.

The ECWS consists of two equally sized divisions. Each division can provide 100 percent of the cooling capacity required to meet system demands during normal and accident conditions. Each division is supplied from independent Class 1E power sources and the respective CCWS trains. The ECWS supplies chilled water to the safety-related HVAC cooling loads in the control room, computer room, electrical rooms, mechanical rooms, subsphere pump rooms, and penetration rooms.

The ECWS is located in a flood-protected and tornado-missile protected seismic Category I structure. The ECWS is designed in accordance with the seismic Category I and Class 1E requirements. The ECWS is protected from pipe breaks, pipe whip, tornado missiles, jet impingement, and severe environmental conditions.

The design of the ECW system complies with GDC 2 and 4 with respect to protection against natural phenomena, internally and externally-generated missiles, and dynamic effects resulting form postulated piping failures. The design also complies with GDC 5, 44, 45, and 46 with respect to shared systems, cooling water requirements, and ISI and testing requirements. Therefore, the staff concludes that the system design meets the applicable acceptance criteria of SRP Section 9.2.2.

The NCWS consists of two equally sized divisions. Each division can provide 100 percent of the cooling capacity required to meet system demands during normal conditions. The NCWS is not safety related because it is not required for ensuring the RCS capability to achieve and maintain safe shutdown, and to prevent or mitigate offsite radiological exposures during accidents. Therefore, GDC 44, 45, and 46, identified as acceptance criteria in SRP Section 9.2.2 for safety-related portions of cooling water systems, are not applicable to the NCWS.

The NCWS complies with GDC 2 with respect to protection of its safety-related portions against natural phenomena and protection of other safety-related systems against the consequences of failure of the non-seismic portions of the system, as specified by SRP Section 9.2.2 acceptance criteria. Therefore, the staff concludes that the NCWS meets the applicable acceptance criteria of SRP Section 9.2.2.

Therefore, Issue 143 is resolved for the System 80+ design.

Issue 153: Loss of Essential Service Water in LWRs

Issue 153, in NUREG-0933, addressed the reliability of ESW systems and related operating problems. In a comprehensive NRC evaluation of operating experience related to ESW systems (NUREG-1275, Volume 3, "Operating Experience Feedback Report," dated November 1988), a total of 980 operational events involving the ESW system were identified, of which 12 resulted in complete loss of the ESW system.

Among the causes of failure and degradation are (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion), (2) ice effects, (3) single failures and other design deficiencies, (4) flooding, (5) multiple equipment failures, and (6) human and procedural errors.

At each plant, the ESW system supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink. The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate the consequences of the accident. Under normal operating conditions, the ESW system provides component and room cooling (mainly via the component CCWS). During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to FPS, cooling towers, and water-treatment systems at a plant.

The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the NSSS. As a result, generic solutions (if needed) are likely to be different for PWRs and BWRs. The possible solutions are (1) installation of a redundant intake structure including a service water pump, (2) hardware changes to the ESW system, (3) installation of a dedicated RCP seal cooling system, or (4) changes to TS or operational procedures.

In the resolution of Issue 130, the staff surveyed seven multiplant sites and found that loss of the ESW system could be a significant contributor to core-damage frequency. The generic safety insights gained from this study supported previous perceptions that ESW system configurations at other multiplant and single-plant sites may also be significant contributors to plant risk and should also be evaluated. As a result, this issue was identified to address all potential causes of ESW system unavailability, except those that had been resolved by implementation of the requirements in GL 89-13.

In CESSAR-DC Section 20.2.48, ABB-CE states that the System 80+ design includes a safety-grade SSWS, with redundant and independent trains (CESSAR-DC, Section 9.2.1). RCP seal cooling is provided by two systems, which are backed up by a dedicated seal injection pump as described in the resolution of Issue 23. TSs for the SSWS are provided in CESSAR-DC, Chapter 16. The staff has evaluated ABB-CE's response and finds it acceptable because this design complies with criteria in NUREG-0933 on this issue. The SSWS is approved in Section 9.2.1 of this report.

Therefore, Issue 153 is resolved for the System 80+ design. See the resolution of Issue 51 on improving SSWS reliability in this section.

Issue 155.1: More Realistic Source Term Assumptions

Issue 155, in NUREG-0933, concerned the resolution of the seven recommendations of the TMI-2 Safety Advisory Board. Issue 155.1 is the resolution of the first recommendation.

During the TMI-2 accident, fission products did not behave as had been predicted by the analytical methods and assumptions used in the licensing process at that time. With the completion of a large number of PRAs since the TMI-2 event, the TMI-2 Safety Advisory Board believed that it should be possible to list accident sequences with chemical conditions similar to TMI-2. Such a listing could serve as a guide as to which accidents might be regarded as hazardous, or less hazardous, relative to the possible escape of iodine and could be useful in the future design of safety features. Since some of the assumptions used for source term considerations at TMI-2 were flawed in this respect, the board recommended that the source term be restated using current scientific knowledge.

This issue is being pursued by the staff as part of comprehensive revisions to 10 CFR Parts 50 and 100 to reflect a better understanding of accident source terms and severe accident insights, as well as evaluate the impact of these phenomena on plant ESFs.

ABB-CE stated in CESSAR-DC Section 20.2.49 that the new radiological source term described in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated June 1992, has been implemented for the System 80+ design and described in CESSAR-DC Appendix 15A. The corresponding environmental qualification is in CESSAR-DC Section 3.11 and the supporting CSS effectiveness analysis is in CESSAR-DC Section 6.5.

The staff has evaluated ABB-CE's response to this issue and finds ABB-CE's implementation of the new source terms acceptable as discussed in Chapter 15, Appendix 15A, and Section 19.2.3.3.6.1 of this report.

Therefore, Issue 155.1 is resolved for the System 80+ design.

20.4 Three Mile Island Action Plan Items

Except for Issue II.E.2.2, the TMI Action Plan items are evaluated against the System 80+ design in this section:

- for the design to comply with 10 CFR 52.47(a)(1)(iv) and 10 CFR 50.34(f)
- because ABB-CE stated in CESSAR-DC Table 20.1-1 that the TMI Action Plan item applied to System 80+

The staff also decided to include a discussion of Issue II.E.2.2 for the System 80+ design.

Issue I.A.1.4: Long-Term Upgrading of Operating Personnel and Staffing

Issue I.A.1.4, in NUREG-0933, addressed changes to 10 CFR 50.54 concerning shift staffing and working hours of licensed operators. ABB-CE considers this issue not relevant to the System 80+ design because it is an operational issue which is outside the scope of System 80+ design certification. The organizational structure of the site operator is discussed in Section 13.1 of this report. The COL applicant will be responsible for addressing this issue as part of the licensing process. This is acceptable and is included in COL Action Item 13.1-1.

Therefore, Issue I.A.1.4 is resolved for the System 80+ design.

Issue I.A.4.1(2): Interim Changes in Training Simulators

Issue I.A.4.1(2), in NUREG-0933, addressed the specific training simulator weaknesses identified in the short-term study of I.A.4.1(1), NUREG/CR-1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," dated August 1980. ABB-CE considers this issue not relevant to the System 80+ design because it is outside the scope of System 80+ design certification. Training materials are discussed in Section 13.2 of this report. The COL applicant will be responsible for addressing this issue as part of the COL process. This is acceptable and included in COL Action Item 13.2-1.

Therefore, this issue is resolved for the System 80+ design.

Issue I.A.4.2: Long Term Training Simulator Upgrade

Issue I.A.4.2, in NUREG-0933, addressed the capabilities of training simulators. This issue was resolved by Revision 1 to RG 1.149 ("Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations"), 10 CFR 55.45(b) on approved or certified simulation facility in licensed operator operating tests, and NUREG-1258 ("Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55," dated December 1987).

The staff considers this an operational issue and, therefore, outside the scope of the System 80+ design. As stated in CESSAR-DC Table 20.1-1 for this issue, ABB-CE also considers this issue not relevant to the System 80+ design because it is an operational issue. The COL applicant will be responsible for providing the site-specific information at the COL phase. This is acceptable and is part of COL Action Item 13.2-1.

Therefore, Issue I.A.4.2 is resolved for the System 80+ design. For DSER Open Item 20.3-1, in a footnote to the table in DSER Section 20.3, the staff stated in part that Issue I.A.4.2 would be evaluated in this report. The preceding discussion on Issue I.A.4.2 resolves this part of the open item.

Issue I.C.1: Guidance for Evaluation and Development of Procedures for Transients and Accidents

Issue I.C.1, of NUREG-0933, addressed the preparation of emergency operating procedures (EOPs). For the System 80+ design, ABB-CE designates these EOGs. The information on EOGs should provide assurance that operator actions are technically correct and the procedures are easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance must be in compliance with the guidance provided in NUREG-0737 and its Supplement 1. The EOGs must be function-oriented procedures to mitigate the consequences of the broad range of mitigating events and subsequent multiple failures or operator errors, without the need to diagnose specific events.

Addressing the concerns in Issue I.C.1, ABB-CE states in CESSAR-DC Section 20.2.88 that the ultimate responsibility of preparing the plant-specific EOPs to be consistent with guidance in NUREG-0737 and its Supplement 1 remains with the COL applicant or the owner/operator. However, by providing EOGs, ABB-CE will assist the owner/operator in preparing the plant-specific EOPs and in training plant operators.

The EOGs in CEN-152 have generic applicability. The guideline structure was designed to accommodate revisions necessary for plant-specific features to ensure operational compatibility. In the response to staff RAI Q440.23, listed in Appendix B to this report, regarding applicability of the existing EOGs to the System 80+ design, ABB-CE stated that the EOGs in CEN-152 are applicable to the System 80+ plant; however, modifications will be made to the EOGs to account for System 80+ design improvements, which include (1) four (instead of two in the existing applicant's plants) high-pressure safety-injection pumps, (2) additional EFW pumps, (3) interchangeability of containment spray and shutdown cooling pumps, (4) IRWST, (5) SDS, (6) cavity flooding system, and (7) AAC power supply. ABB-CE committed to modify the EOGs within the then-current CEN-152 structures to ensure operational compatibility with the System 80+ design and to include an appropriate analytical basis.

ABB-CE developed the EOGs to satisfy the requirements of Issue I.C.1. The staff previously reviewed the EOGs in CEN-152 and issued SERs for approval of Revisions 1 and 2 and Submittals 1 and 2 to Revision 3 of CEN-152 on July 29, 1983; April 16, and November 7, 1985; and November 5, 1986, respectively. The current EOGs are a revision of Submittal 2 to Revision 3 of CEN-152. They retain the overall technical content of the existing CE-EOGs and include improvements identified through user review and previous staff review comments in the issued SERs. In the letter dated August 2, 1988, the staff allowed implementation of the current CEN-152 to improve the plant-specific EOGs. With the commitment of ABB-CE to modify the current CEN-152 to be compatible with the System 80+ design, the staff determined that ABB-CE will prepare adequate EOGs to satisfy guidance in NUREG-0737 and its Supplement 1, pending staff acceptance of the updated EOGs that ABB-CE will submit. This was designated Confirmatory Item 20.2-3 in the DSER.

In response to DSER Confirmatory Item 20.2-3, ABB-CE submitted the System 80+ EOGs, in Attachment 4 to its letter dated January 18, 1993, for staff review. These EOGs are a revision of the latest version of CEN-152 and reflect the design features of the System 80+ design. ABB-CE also submitted a "deviation" document identifying the procedural difference from CEN-152 with supplemental information (ABB-CE letter dated April 1, 1993) to explain the technical bases for the deviations. During the review, in a letter dated September 1, 1993, ABB-CE addressed the staff RAI's Q440.223 through Q440.246, and submitted additional EOG changes to resolve open items identified in the DSER. The staff has reviewed the EOGs for the System 80+ design, the deviation document, the responses to the staff RAIs, and EOG changes for closure of open items and concludes that the EOGs are adequate for the System 80+ design and acceptable for the following reasons:

- The System 80+ EOGs retain the structure and event mitigation strategies presented in CEN-152 and contain both symptom-oriented and function-based procedural guidelines. The symptom-oriented procedural guidelines include the guidance for standard post trip actions, reactor trip recovery, excess steam demand, LOCA, loss of offsite ac power, total loss of feedwater, SGTR, and SBO. The function-based recovery guidelines address such safety functions as reactivity control, maintenance of vital power sources, reactor inventory and pressure control, RCS and core heat removal, containment temperature and pressure control, containment isolation, and containment combustible gas control.

- The System 80+ EOGs were modified to reflect the System 80+ design, including such design features as four SI pumps (instead of two high pressure and two low-pressure SI pumps in the existing ABB-CE plants), additional EFW pumps, interchangebility of containment spray and shutdown cooling pumps, IRWSTs, AAC power supply, and SDS.

- The System 80+ EOGs adequately incorporate the procedural guidelines required for resolving the open items in the DSER. The EOG changes for the resolution of open items are:

  - SI flow rate at the low-pressure range (See Section 6.3.1 of this report for resolution of DSER Open Item 6.3.1-1).

  - Use of the reactor coolant gas vent (RCGV) system for RCS pressure control (See Section 6.7.1 of this report for resolution of DSER Open Item 6.7.1-1).

  - Use of the RDS for the feed-and-bleed operation (See Section 6.7.2 of this report for resolution of DSER Open Item 6.7.2-4).

  - Procedure changes reducing challenges to the primary safety valves to open during an SGTR event (See Section 15.3.9 of this report for resolution of DSER Open Item 15.3.8-1).

- Avoidance of deboration during an SGTR event (See Section 15.3.9 of this report for resolution of DSER Open Item 15.3.8-2).

- Use of a DSIS for RCP seal cooling (See Issue 23 in Section 20.3 of this chapter for resolution of DSER Open Item 20.2-7).

Since ABB-CE submitted adequate EOGs for the System 80+ design, the staff concludes that DSER Confirmatory Item 20.2-3 is closed.

In DSER Open Item 20.2-21, the staff stated that the COL applicant will be responsible for submitting plant-specific EOPs to comply with NUREG-0737 and its Supplement 1. ABB-CE has included the site-specific procedure development process as a COL action item in CESSAR-DC Section 13.5.1 and states that the methods and criteria for the development, validation and verification, implementation, maintenance, and revision of procedures will include considerations of Issue I.C.1. As discussed in Section 13.5 of this report, this is in COL Action Item 13.5-1. Therefore, DSER Open Item 20.2-21 is resolved.

Therefore, Issue I.C.1 is resolved for the System 80+ design.

Issue I.C.5: Procedures for Feedback of Operating Experience to Plant Staff

Issue I.C.5, in NUREG-0933, addressed the quality of procedures for feedback of experience at operating plants. This issue was clarified in NUREG-0737 and requirements were issued there. Development of detailed procedures is outside the scope of System 80+ design certification and is the responsibility of the COL applicant. ABB-CE has included the procedure development process as a COL action item in CESSAR-DC Section 13.5.1, as discussed in Issue I.C.1 above.

The COL applicant will be responsible for the site-specific information at the COL phase. In CESSAR-DC Section 13.5.1, ABB-CE states that the methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures will include considerations of Issue I.C.5. The COL applicant will be responsible for providing the site-specific information at the COL phase. In Section 13.5 of this report, the staff designates this as COL Action Item 13.5-1 and concludes it is acceptable.

Therefore, Issue I.C.5 is resolved for the System 80+ design.

Issue I.C.9: Long-Term Program for Upgrading Procedures

Issue I.C.9, in NUREG-0933, addressed the upgrading of procedures at operating plants. With the exception of EOPs, this issue was clarified in Supplement 1 of NUREG-0737 and resolved with Revision 1 of SRP Section 13.5.2. The EOPs are handled through the resolution of Issue I.C.1.

For this issue, the staff determined that development of detailed procedures is outside the scope of System 80+ design certification and is the responsibility of the COL applicant. ABB-CE has included the procedure development process as a COL action item in CESSAR-DC Section 13.5.1. In CESSAR-DC Section 13.5.1, ABB-CE states that the methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures will include considerations of Issue I.C.9.

The COL applicant will be responsible for providing the site-specific information at the COL phase as discussed in Section 13.5 of this report. This is part of COL Action Item 13.5-1 and is acceptable.

Therefore, Issue I.C.9 is resolved for the System 80+ design. See also the resolution of Issue HF4.4 in Section 20.5 of this report and of Issues I.C.1 and I.C.5 in this section.

Issue I.D.1: Control Room Design Reviews

Issue I.D.1, in NUREG-0933, addressed licensees performing a detailed review of their control room using human factors engineering (HFE) techniques and guidelines to identify and correct design deficiencies. This issue was clarified in NUREG-0737 and NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981, and is considered resolved. See also Issue I.D.4 in this section.

ABB-CE states in CESSAR-DC Section 20.2.90 that the discussion on HFE in Chapter 18 of CESSAR-DC summarizes this issue. Chapter 18 of this report evaluates the System 80+ HFE, including its application to the control room. In Section 18.10, the staff concludes that the HFE program is acceptable and that the program supplies an acceptable framework for developing human factors interfaces of the control room. The staff reviewed basic design features of the control room and found them consistent with human factors standards, guidelines, and principles, and acceptable for use in the control room. All previously identified DSER issues in Chapter 18 are resolved.

Therefore, Issue I.D.1 is resolved for the System 80+ design.

Issue I.D.2: Plant Safety Parameter Display Console

Issue I.D.2, in NUREG-0933, addressed improving the presentation of the information provided to control room operators. The requirements for this issue are in Supplement 1 to NUREG-0737. This issue raised the need for a SPDS that clearly displays a minimum set of parameters defining the safety status of the plant. Paragraph (2)(iv) of 10 CFR 50.34(f) requires a plant SPDS console that will provide such a display to operators, and that is capable of displaying a full range of important plant parameters and data trends on demand and indicating when process limits are being approached or exceeded.

In CESSAR-DC Sections 18.7.1.8.1 and 20.2.91 and the revised OER, ABB-CE indicates how the System 80+ design complies with the SPDS criteria. The staff reviewed the control room design according to the SPDS criteria in Supplement 1 to NUREG-0737 and found it acceptable. This review is described in greater detail in Sections 18.3.3.2.5 and 18.6.1.3.1.4 of this report. In these sections, the staff finds that the responses and commitments of ABB-CE regarding the eight SPDS requirements of Supplement 1 to NUREG-0737 are acceptable.

Therefore, Issue I.D.2 is resolved for the System 80+ design.

Issue I.D.3: Safety System Status Monitoring

Issue I.D.3, in NUREG-0933, addressed the need for those licensees and applicants who have not committed to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to install a bypass and inoperable status indication system to give operators timely information on the status of the safety systems.

Resolution of this issue requires adoption of the guidelines in RG 1.47. In CESSAR-DC Section 20.2.92, ABB-CE commits to this RG. The commitment of ABB-CE to these guidelines is evaluated in Section 7.1 of this report and the staff found ABB-CE's method for implementing RG 1.47 acceptable.

Therefore, Issue I.D.3 is resolved for the System 80+ design.

Issue I.D.4: Control Room Design Standard

Issue I.D.4, in NUREG-0933, addressed the need for guidance on the design of control rooms to incorporate human factors considerations and the desirability of endorsing an industry standard for future control room designs.

Under Issue I.D.1 in this section, the NRC issued NUREG-0700 as guidance for detailed control room design reviews to conform to accepted human factors principles. The staff issued SRP Section 18.1, to document the NRC review process for control room designs, and Appendix B of NUREG-0700, to provide guidance for designing new control rooms.

By letter dated December 18, 1992, and CESSAR-DC Section 20.2.43, ABB-CE indicates that this issue is resolved because (1) the System 80+ human factors program is being conducted in accordance with a HFE program plan, which is based on current HFE program guidance; and (2) the System 80+ human factors program is governed by the use of HFE standards, guidelines, and bases documents, which have been approved by the staff. The HFE program plan is evaluated in Section 18.3 of this report. ABB-CE states that the advanced control room design will meet the applicable regulations related to appropriate HFE principles through the implementation of the human factors program plan, including the use of detailed verification and validation methods.

Therefore, Issue I.D.4 is resolved for the System 80+ design.

Issue I.D.5(1): Control Room Design - Improved Instrumentation Research Alarms and Displays

Issue I.D.5(1), in NUREG-0933, addressed the man-machine interface in the control room with regard to the use of lights, alarms, and annunciators to reduce the potential for operator error, information overload, unwanted distractions, and insufficient information organization.

ABB-CE determined lighting and illumination levels in the "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" which is presented in CESSAR-DC Section 18.6 and evaluated in Section 18.3.3.2.5 of this report for Issue I.D.5(1). The technical adequacy of the aforementioned document was found acceptable. The staff evaluated the System 80+ annunciator and alarm systems during the onsite design features evaluation. The staff concluded in its letter dated July 15, 1993, which provided the minutes of the May 13 and 14, 1993, public meeting, that for the most part these systems were acceptable except for some issues that were raised. The staff indicated in the public meeting minutes, that issues could be solved by a commitment of ABB-CE to incorporate the issues into its HFE tracking system. ABB-CE agreed to address the staff's specific concerns through evaluation and resolution of specific alarm system issues in the tracking system. As documented in Attachment 3 to the ABB-CE letter dated January 7, 1994, Issue No. 101, which provides a commitment for prototype testing and a number of prior items that provide for tracking of the May meeting issues, has been included in the tracking system. This resolves Confirmatory Item 20.2-1 which had been identified in the advanced copy of this report.

Therefore, Issue I.D.5(1) is resolved for the System 80+ design.

In the DSER, the staff stated that Issue I.D.5(1) will be discussed in this report and designated this as DSER Open Item 20.2-24. The staff's evaluation of this issue is given above. Therefore, DSER Open Item 20.2-24 is resolved.

Issue I.D.5(2): Control Room Design - Improved Instrumentation Research — Plant Status and Postaccident Monitoring

Issue I.D.5(2), in NUREG-0933, addressed the need to improve the operators' ability to prevent, diagnose, and properly respond to accidents. This issue was originally raised in 1980, in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated May 1980, and led to new NRC requirements. Guidance for addressing the issue is in RG 1.47, which describes an acceptable method for implementing the requirements of IEEE 279-1971 ("Criteria for Protection Systems for Nuclear Power Generating Stations") and Appendix B (Criterion XIV) of 10 CFR Part 50, with respect to the bypass or inoperable status of safety systems; and RG 1.97, which defines an acceptable method for implementing NRC requirements to provide instrumentation and to monitor plant variables and systems during and following an accident.

The acceptance criteria for the resolution of this issue are:

- For ESF status monitoring, RG 1.47 recommends automatic bypassed or inoperable status indication at the system level for plant protection systems, safety systems actuated or controlled by protection systems, and their auxiliary and supporting systems. These features should indicate in the control room and should have manual input capability.
- For PAMI, RG 1.97 gives criteria for design and qualification of the instrumentation. Three categories (designated 1, 2, and 3) provide a graded approach to requirements based on the importance to safety of the variable being monitored. Criteria exist for equipment qualification, redundancy, power sources, channel availability, QA, display and recording range, equipment identification, interfaces, servicing, testing and calibration, human factors, and direct measurement. The actual variables to be monitored are tabulated by type, and the instrumentation design and qualification requirement (Category 1, 2, or 3) is identified for each variable.

ABB-CE addresses the resolution of this issue for the System 80+ design in CESSAR-DC Section 20.2.95. The System 80+ design provides bypassed or inoperable status indication for the RPS, ESFAS, the systems they control, and their auxiliary or support systems. The functional design requirements conform with RGs 1.47 and 1.97 and, therefore, are acceptable. The staff will address implementation of these functional design requirements during its review of the ITAACs for software systems implementation, which is discussed in Section 14.3 of this report.

Therefore, Issue I.D.5(2) is resolved for the System 80+ design.

Issue I.D.5(3): Control Room Design - On-Line Reactor Surveillance Systems

Issue I.D.5(3), in NUREG-0933, addressed the benefit to plant safety and operations of continuous on-line automated surveillance systems. Systems that automatically monitor reactor performance can benefit plant operations and safety by providing continuous diagnostic information to the control room operators, to predict anomalous plant behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise-monitoring in BWRs to detect vibrations in internal components, and pressure noise surveillance at TMI-2 to monitor primary loop degasification. On-line surveillance data have been used to assess loose thermal shields.

ABB-CE states in CESSAR-DC Section 20.2.96 that continuous on-line surveillance of the NSSS involves the following areas for which acceptance criteria are separately defined:

- vibration monitoring of reactor internals
- RCPB leakage detection
- loose-parts monitoring

The acceptance criteria for the resolution of Issue I.D.5(3) for monitoring vibrations in internal components are in ANSI/ASME OM-5-1981, "Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors." This standard makes recommendations on the use of ex-core neutron detector signals for monitoring core barrel axial preload loss. This standard also documents a program containing baseline, surveillance, and diagnostic phases and makes recommendations for data acquisition frequency and analysis.

The acceptance criteria for leak monitoring are in RG 1.45 that documents acceptable methods for channel separation, leakage detection, detection sensitivity and response time, signal calibration, and seismic qualification of RCPB leakage detection systems. It defines the regulatory position for an acceptable design of these systems.

The acceptance criteria for loose-parts monitoring are in RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." This RG gives guidelines on such system characteristics as sensitivity, channel separation, data acquisition, and seismic and environmental conditions for operability. It also identifies alert levels, data acquisition modes, safety analysis reports, and TS pertaining to a LPMS.

The System 80+ design incorporates an NIMS that detects deterioration of the NSSS pressure boundary. The system is described in Section 7.7.1.20 of this report and consists of the IVMS, ALMS, and LPMS subsystems, that are also discussed in Issue C-12 of Section 20.2 of this report. In Section 7.7.2 of this report, the staff concludes that the functional designs of these control systems are acceptable. This conclusion applies to all portions of these systems except for the items that apply to digital systems, as stated in Section 7.1 of this report.

Therefore, Issue I.D.3(5) is resolved for the System 80+ design.

Issue I.D.5(4): Improved Control Room Instrumentation: Process Monitoring Instrumentation

Issue I.D.5(4), in NUREG-0933, addressed the benefit to plant safety and operations of improved measurement of certain reactor parameters (e.g., reactor vessel water level and relief valve flow), and of parameters when they are outside their normal operating range.

The TMI-2 accident identified the need to improve process monitoring instrumentation. As a result, new and improved monitoring systems, such as inadequate core cooling (ICC) instrumentation, extended-range postaccident monitoring of selected reactor parameters, reactor-vessel-level monitoring systems (RVLMS), and monitoring systems for detecting primary pressure boundary leakage, were developed.

The acceptance criteria for the resolution of Issue I.D.5(4), improving process instrumentation, are in NUREG-0660 and NUREG-0737. TMI Action Plan Item II.F.2 of NUREG-0737 gives requirements for the design of instrumentation that detects conditions leading to ICC. TMI Action Plan Item II.D.3 of NUREG-0737 gives requirements on direct indication of relief and safety valve position.

ABB-CE discusses its resolution of this issue for the System 80+ design in CESSAR-DC Section 20.2.97. The System 80+ design includes ICC monitoring instrumentation and the instrumentation is acceptable, as discussed in Section 7.5 of this report.

The acceptance criteria for the extended-range sensors are in RG 1.97 in a tabulation of acceptable ranges for the PAMI. The type and range of the System 80+ sensors are acceptable, as discussed in Section 7.5 of this report.

Therefore, Issue I.D.5(4) is resolved for the System 80+ design.

Issue I.F.1: Expanded Quality Assurance List

Issue I.F.1, in NUREG-0933, addressed improving the QA program for the design, construction, and operation of nuclear power plants. The licensees were to identify those SSCs that were not labeled safety-related but were important to safety, to prioritize their importance to safety, and to prepare a generic QA list. In GDC 1, the NRC requires that SSCs important to safety should be designed, built, and tested commensurate to their importance to safety. In January 1984, the staff issued GL 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety-Related'," dated January 5, 1984, to clarify the use of the terms "important to safety" and "safety-related."

ABB-CE states in CESSAR-DC Section 20.2.98 that the classification of SSCs for the System 80+ design is provided in CESSAR-DC Table 3.2-1. The staff evaluates this table in Section 3.2 of this report and concludes that it conforms to GDC 1.

Therefore, Issue I.F.1 is resolved for the System 80+ design.

Issue I.F.2: Develop More Detailed Quality Assurance Criteria

Issue I.F.2, in NUREG-0933, addressed improvements to the QA program for the design, construction, and operation at nuclear power plants to provide greater assurance that these activities are conducted in a manner commensurate with their importance to safety. The subissues for Issue I.F.2 that must be addressed for 10 CFR 52.47(a)(1)(iv) are the following: Item 2, include QA personnel in review and approval of plant procedures; Item 3, include QA personnel in all design, construction, installation, testing, and operation activities; Item 6, increase the size of the QA staff; and Item 9, clarify organizational reporting levels for the QA organization. The new requirements were incorporated into the SRP (third edition) on QA.

ABB-CE states in CESSAR-DC Section 20.2.99 that the QA program for the System 80+ is described in CESSAR-DC Section 17.1 and approved by the staff. The staff's approval of this program is discussed in Section 17.1 of this report.

ABB-DC has only addressed the QA program for the design of System 80+. The QA program for the COL applicant's design, construction, and operation phases are outside the scope of System 80+ design certification and are designated COL Action Items 17.1-1 and 17.2-1 in Chapter 17 of this report. The COL applicant will have the responsibility of addressing this issue for the design of the remaining parts of the plant, and for the modification and operation of the plant.

Therefore, Issue I.F.2 is resolved for the System 80+ design.

Issue I.G.2: Scope of Test Program

Issue I.G.2, in NUREG-0933, addressed the need for licensees to develop a more comprehensive preoperational and low-power test program for their plant to find any anomalies in the response of the plant to a transient during the initial test program (ITP). With the revisions to the SRP and the NRC Office of Inspection and Enforcement Manual (June 1989 revision to NUREG-0933), this issue was considered resolved.

In CESSAR-DC Section 20.2.100, ABB-CE refers to the startup (or initial) test program in Chapter 14 of CESSAR-DC. In Section 14.2 of this report, the staff evaluates the ITP for the System 80+ design. The staff reviewed the scope of the test program and concludes the test abstracts and acceptance criteria are acceptable. ABB-CE described the typical licensee's organization and staffing for this design which the staff found acceptable; the COL applicant is responsible for developing the specific organization and staffing levels appropriate for its facility. ABB-CE also described the methods the COL applicant can use for preparation and organization approval of Phase I through Phase IV test procedures in SRP Section 14.2 ("Initial Plant Test Program - Final System Analysis Report"). The COL applicant has the responsibility for the preparation and organization approval of these procedures. These are designated COL Action Items 14.2.3-1 through 14.2.3-4.

In Section 14.2 of this report, the staff concludes that the ITP for the System 80+ design is acceptable; therefore, Issue I.G.2 is resolved for the System 80+ design.

Issue II.B.1: Reactor Coolant System Vents

Issue II.B.1, in NUREG-0933, addressed the requirements in 10 CFR Part 50 and NUREG-0737 to install reactor vessel and RCS high-point vents. These vents are designed to release non-condensable gases from the RCS to avoid loss of core cooling during natural circulation. The design of these vents must conform to the applicable GDC requirements of 10 CFR Part 50 (Appendix A), which are listed below, and meet the applicable codes and standards for the RCS pressure boundary.

ABB-CE states in CESSAR-DC Section 20.2.101 that the System 80+ design includes a SDS that performs the RCGV function to meet the requirements in NUREG-0737. The RCGV system is described in CESSAR-DC Section 6.7.1.2.2. The staff has reviewed the RCGV system design and concludes, in Section 6.7.1 of this report, that it is acceptable because it meets the following design criteria: (1) the system must be operable from the control room (GDC 19), (2) the system must be testable (GDC 36), (3) the system must be capable of functioning following a LOOP (GDC 17), and (4) the system must be able to withstand an OBE (RG 1.29).

Therefore, Issue II.B.1 is resolved for the System 80+ design.

Issue II.B.2: Safety Review Consideration - Plant Shielding To Provide Postaccident Access to Vital Areas

Issue II.B.2, in NUREG-0933, addressed having licensees perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The review would locate vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, where occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems. The requirements were given in NUREG-0737 and the issue was resolved.

ABB-CE states in CESSAR-DC Section 20.2.102 that a radiation and shielding design will be reviewed during the detailed design phase of the plant and referred to CESSAR-DC Sections 12.2.3 (post-accident sources), 12.3.1.3 (vital areas for post-accident access), and 3.11 (environmental qualification of equipment). These sections are evaluated in Sections 12.2.3, 12.3.1 and 12.3.2, and 3.11 of this report, respectively, and are accepted by the staff.

The COL applicant will be responsible for the detailed shielding design review of the plant as discussed in Section 12.1.2 of this report. The completion of this review and the submittal of the review to the staff is included in COL Action Item 12.1.2-1.

Therefore, Issue II.B.2 is resolved for the System 80+ design.

Issue II.B.3: Postaccident Sampling Capability

Issue II.B.3, in NUREG-0933, addressed upgrading postaccident sampling at plants. The requirements are in NUREG-0737. The reactor coolant and containment atmosphere sampling-line systems should permit personnel to take a sample under accident conditions promptly and safely. The radiological spectrum analysis facilities should be capable of quantifying certain radionuclides that are indicators of the degree of core damage promptly. In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions.

ABB-CE indicates in CESSAR-DC Section 20.2.103 that the System 80+ design includes a PSS which permits sampling during reactor operation, cooldown, and post-accident conditions without requiring access to the containment. The staff's evaluation is in Section 9.3.2 of this report. As discussed in this section, the Commission approved exemptions so that the capabilities of the postaccident sampling system for the design does not include the determination of the hydrogen concentration in the containment atmosphere or chlorides in the reactor coolant, and has the time limit for analysis of the reactor coolant boron and radioactivity concentration of 8 and 24 hours, respectively. The staff concludes that ABB-CE has adequately addressed postaccident sampling.

Therefore, Issue II.B.3 is resolved for the System 80+ design.

Issue II.B.8: Rulemaking Proceedings on Degraded Core Accidents Description

Issue II.B.8, in NUREG-0933, addressed degraded core accidents discussed in safety reviews of the plant. The work on hydrogen control led to the hydrogen control rule that was approved by the Commission and published in the Federal Register on January 25, 1985. The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than DBA (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070, "NRC Policy on Future Reactor Designs," dated July 1985. Thus, with the issuance of the rule on hydrogen control, this item was resolved and new requirements were established.

ABB-CE states in CESSAR-DC Section 20.2.104 that the analysis of degraded core conditions and the capability of System 80+ to mitigate those conditions is provided in Section 19.11 of CESSAR-DC. In response to this issue, applicants were expected to address the feasibility of mitigating features arising from severe accident considerations, including the conduct of conceptual designs for filtered and vented containment, core-retention devices, and hydrogen control systems. The requirements for such analyses were subsumed into 10 CFR 50.34(f)(1)(i). This section of the regulations specifically requires all applicants (for a LWR CP or manufacturing license) to perform a plant/site-specific PRA, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. 10 CFR 52.47(a)(1)(v) also requires that applications for standard design certifications contain a design-specific PRA.

Consistent with the above requirements, ABB-CE has submitted a design-specific PRA for the System 80+ design and has used the PRA as the basis for evaluating potential design improvements in the reliability of core and containment heat removal systems. The PRA and ABB-CE's evaluation of potential design enhancements is documented in CESSAR-DC Chapter 19. The staff's evaluation of the PRA and potential System 80+ design enhancements is provided in Section 19.1 of this report.

On this basis, Issue II.B.8 is resolved for the System 80+ design.

Issue II.C.4: Reliability Engineering

Issue II.C.4, in NUREG-0933, addressed improving system-oriented approaches to safety reviews, specifically in this case reliability engineering or assurance; however, no requirements exist for licensees to have a reliability assurance program (RAP). Such programs determine system availabilities, identify high component failure rates, determine root causes for component failures, identify possible corrective actions, and perform what is generally called reliability engineering. This issue was raised to determine what licensees were doing to implement a RAP. This issue was resolved with no new requirements.

ABB-CE states in CESSAR-DC Section 20.2.105 that a PRA has been performed for the System 80+ design and the COL applicant will be required to implement an operability assurance program to ensure that the PRA remains valid during the plant operation. This does not fully address Issue II.C.4 for the System 80+ design; however, ABB-CE discusses the RAP for the design phase of System 80+ in Section 17.3 of CESSAR-DC.

The ABB-CE RAP for the design phase of the System 80+ design is evaluated in Section 17.3 of this report. In that section, there are actions to be performed by the COL applicant which are designated COL Action Items 17.3.1-1 and 17.3.9-1. The COL applicant's RAP is discussed in Section 17.3.10 of this report. Based on Section 17.3 of this report, the staff concludes that the RAP for the design phase of System 80+ is acceptable.

Therefore, Issue II.C.4 is resolved for the System 80+ design. See also Issue 23 in Section 20.3 of this report for additional discussion on the RAP.

Issue II.D.1: Performance Testing of PWR Safety and Relief Valves

Issue II.D.1, in NUREG-0933, addressed the requirements in NUREG-0737 for qualification testing of RCS safety, relief, and block valves under expected operating conditions for design-basis transients and accidents, including ATWS. This issue was resolved by requiring licensees to conduct testing to qualify reactor coolant relief valves, safety valves, block valves, and associated discharge piping.

ABB-CE states in CESSAR-DC Section 20.2.106 that the System 80+ design uses the pressurizer safety valves instead of PORVs to protect the RCS from overpressurization. PORVs and block valves are not used in the System 80+ design. See Issue 70 in Section 20.3 of this chapter.

The safety valve test program conforms to the requirements in NUREG-0737 and includes adjustments to valve ring setting combinations to provide stable valve operation using the EPRI safety valve test program findings documented in CEN-227, "Summary Report on the Operability of Pressurizer Safety Relief Valves in CE Designed Plants."

Although NUREG-0737 specifies that ATWS is to be considered in developing test conditions, ABB-CE maintains that it need not consider ATWS conditions in the testing program because the design employs an APS that uses an independent and diverse control-grade reactor trip and turbine trip specifically designed to address prevention of ATWS events. The staff's evaluation of the APS is in Sections 7.7.1.12, 7.2, and 15.3.10 of this report. The APS uses an ARTS off pressurizer pressure for the prevention of an ATWS, instead of relief valves for the mitigation of an ATWS. Thus, the reason that ABB-CE did not consider ATWS in its testing program is acceptable.

Therefore, Issue II.D.1 is resolved for the System 80+ design. See also Issue A-9 in Section 20.2 of this report.

Issue II.D.3: Coolant System Valves - Valve Position Indication

Issue II.D.3, in NUREG-0933, addressed the requirements in NUREG-0737 for positive indication in the control room of RCS relief or safety valve position. The acceptance criterion for the resolution of this issue is that the plant design shall include safety and relief valve indication derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe in accordance with the requirements in NUREG-0737. This indication shall have the following design features:

- Unambiguous safety and relief valve indication shall be provided to the control room operator.
- Valve position should be indicated within the control room and should be alarmed.
- Valve position indication may be either safety or control grade; if it is control grade, it must be powered

from a reliable (e.g., battery-backed) instrument bus (see RG 1.97).

- Valve position indication should be seismically qualified consistent with the component or system to which it is attached.
- Valve position indication shall be qualified for the appropriate operating environment which includes the expected normal containment environment and an OBE.
- Valve position indication shall be human-factors engineered.

ABB-CE states in CESSAR-DC Section 20.2.107 that the System 80+ standard design incorporates four primary safety valves. The valve discharge is headered and routed to the IRWST. These valves are monitored by three methods described in CESSAR-DC Section 5.2.5.1.2.1:

(1) Positive indication of safety valve position is supplied in the control room by the acoustic leakage monitoring system (ALMS).

(2) Each safety valve is monitored for seat leakage by an in-line resistance temperature detector (RTD), which is located in the discharge line upstream of the discharge header for the safety valves, with an indication in the control room.

(3) Safety valve leakage is indirectly monitored from the safety-grade pressurizer pressure and level instrumentation system also located in the control room. This instrumentation is discussed in connection with EOPs. (See Issue I.C.1 in this section.)

The ALMS is part of the NIMS and is described in Section 7.7 of this report. The function of the ALMS is to detect a leak at specific locations or within specific components in the primary system, including the primary safety valves. The ALMS offers the control room operator a direct and unambiguous method of determining the position (open or closed) of the pressurizer safety valves.

The ALMS is composed of sensors (accelerometers) that are installed on the pressurizer safety valve discharge lines (one per safety valve). Signals from the sensors are routed to the control room. Within the alarm instrumentation, the signal is compared to a threshold value obtained during startup testing.

Alarms are provided as part of the "human engineered" advanced control and are included in the plant computer annunciator systems. After passing through the alarm unit, the amplified accelerometer signals are transmitted to a computer for further analyses. The computer stores, compares trends, and analyzes the data to improve the signal characteristics.

In summary, by providing a direct method for monitoring safety valve position, the ALMS implements the requirements in NUREG-0737 and Issue II.D.3 is resolved for the System 80+ design.

Issue II.E.1.1: Auxiliary Feedwater System Evaluation

Issue II.E.1.1, in NUREG-0933, addressed improving the reliability of the auxiliary feedwater, or EFW, system. The issue addressed the following requirements in NUREG-0737: (1) perform a simplified EFW system reliability analysis to determine the potential for system failure under various loss-of-main-feedwater transients, (2) the acceptance criteria in SRP 10.4.9 and BTP ASB 10-1, and (3) evaluated EFW flow rate design basis and criteria.

ABB-CE states in CESSAR-DC Section 20.2.108 that the System 80+ design has an EFWS to provide reliable and independent safety-related means of supplying secondary-side, quality feedwater to the SGs for the removal of heat and prevention of core uncovery during emergency phases of plant operation, including accidents. The EFWS is a dedicated safety-related system that is not used during normal plant operation. The system is described in Section 10.4.9 of CESSAR-DC.

On the basis of the staff's evaluation of the EFWS in Section 10.4.9 of this report, Issue II.E.1.1 is resolved for the System 80+ design.

Issue II.E.1.2: Auxiliary Feedwater System Automatic Initiation and Flow Indication

Issue II.E.1.2, in NUREG-0933, addressed improving the reliability of the auxiliary feedwater, or EFW, system. It addressed the requirement in NUREG-0737 for plants to install a control-grade system for automatic initiation of the EFW system. The acceptance criteria are in NUREG-0737 and in the design requirements of IEEE 279-1971. Specifically, the system shall incorporate such design features as automatic system initiation, protection from single failure, and environmental and seismic equipment qualification.

ABB-CE states in CESSAR-DC Section 20.2.109 that the System 80+ design uses a dedicated EFWS to provide an independent safety-related means of supplying secondary-side quality feedwater to the steam generator(s) for removal of heat during emergency phases of plant operation. The system is described in CESSAR-DC Section 10.4.9 and evaluated in Section 7.3.9.5 of this report. The EFWS is not used for normal plant operation.

In addition, the EFWS I&Cs are part of the ESF systems and are subject to the design bases described in Section 7.3.9 of this report. These design bases address the applicable GDC identified in 10 CFR Part 50 (Appendix A), including GDC 20.

The EFWS is actuated automatically by an EFW actuation signal (EFAS from the ESFAS or by the APS), which is described in Section 7.7 of this report. In addition to this automatic feature, the EFWS can be manually initiated.

The EFWS, including its integral I&Cs, fulfills the applicable requirements in NUREG-0737 and the design criteria in IEEE 279-1971. The system is evaluated and accepted in Section 10.4.9 of this report.

Therefore, Issue II.E.1.2 is resolved for the System 80+ design. See also Issues 93, 124, and 125.II.7 in Section 20.3 of this chapter.

Issue II.E.1.3: Updated Standard Review Plan and Development of Regulatory Guide

Issue II.E.1.3, in NUREG-0933, addressed improving the reliability of the auxiliary feedwater, or EFW, system. Section 10.4 of the SRP was to be updated, and RG 1.26 was to be revised to include these systems and possibly endorse certain standards. The SRP section was updated in July 1981; however, no additional public and occupational risk reduction was identified to support the need to revise the regulatory guide and it was not revised. This issue is resolved and the requirements were established in the changes to the SRP.

ABB-CE did not consider this issue relevant to the System 80+ design because it was an NRC internal issue to update SRP Section 10.4.9. Although the resolution of this issue was for NRC to update the SRP section, the application of the SRP section to a specific plant design is relevant to the System 80+ design because NUREG-0933 stated, for Issue II.E.1.3, that the resolution of the issue resulted in new requirements and the requirements apply to all PWRs, including the System 80+ design.

In Issues II.E.1.1 and II.E.1.2, ABB-CE discussed the EFW system for the System 80+ design. The staff evaluates the EFWS against SRP Section 10.4.9 in Section 10.4.9 of this report and accepts the design.

Therefore, Issue II.E.1.3 is resolved for the System 80+ design.

Issue II.E.2.2: Research on Small Break LOCAs and Anomalous Transients

Issue II.E.2.2, in NUREG-0933, addressed the NRC research programs focused on small breaks LOCAs (ISLOCAs) and reactor transients. The programs included experimental research in the loss of flow tests (LOFT), semiscale LOFT, Babcock and Wilcox integral system test facilities, systems engineering, and materials effects programs, as well as analytical methods development and assessments in the code-development program.

The programs called for by this issue were completed by the NRC and showed that ECCSs will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single-failure criteria of 10 CFR Part 50 (Appendix K). The application of the experimental data from the research programs to validate the conservatism of the licensing codes used for SBLOCAs are addressed in Issue II.K.3(30) in this section. ABB-CE used the Appendix K criteria and analyses as discussed in CESSAR-DC Section 6.3.3.3 for the System 80+ design as discussed in Chapter 15 of this report.

Therefore, Issue II.E.2.2 is resolved for the System 80+ design.

Issue II.E.3.1: Pressurizer Heater Power Supply

Issue II.E.3.1, in NUREG-0933, addressed requiring that (1) emergency power be available to a minimum number of pressurizer heaters to ensure that natural circulation can be maintained in the RCS if offsite power is lost, (s) establish procedures and training for maintaining the RCS at hot standby conditions with only onsite power available, (3) the time required to connect the heaters to the emergency buses shall be consistent with timely initiation of natural circulation, and (4) pressurizer heater motive and control power shall interface with emergency buses through qualified devices.

ABB-CE states in CESSAR-DC Section 20.2.110 that although no credit is taken for pressurizer heaters to maintain natural circulation if offsite power is lost, the System 80+ design includes two backup pressurizer heater groups, each rated at 200 kW. These heaters are connected to separate 480-V Class 1E buses that are energized from separate and independent EDGs upon the LOOP. Because the heaters are not Class 1E, they are connected to the Class 1E buses through two breakers in series. The criteria for this power supply for pressurizer heaters are consistent with NUREG-0737 requirements and, therefore, acceptable.

Therefore, Issue II.E.3.1 is resolved for the System 80+ design, because the heaters are not needed to maintain natural circulation.

Issue II.E.4.1: Dedicated Hydrogen Penetrations

Issue II.E.4.1, in NUREG-0933, addressed improving the reliability and capability of containment structures to reduce the radiological consequences to the public from accidents, including degraded core events. The issue addressed the need for dedicated containment penetrations for external recombiners and containment purge systems. Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should have containment penetration systems for external recombiner or purge systems that are dedicated to that service only, meet the redundancy and single-failure requirements of GDC 54 and 56, and are of proper size to satisfy the flow requirements of the recombiner or purge system. The requirements in NUREG-0737 that are applicable to the System 80+ design follow:

- An acceptable alternative to the dedicated penetration is a combined design that is single-failure-proof for containment isolation purposes and single-failure-proof for operation of the recombiner or purge system.

- The dedicated penetration or the combined single-failure-proof alternative shall be of proper size to satisfy the flow requirements for the use of the external recombiner or purge system. The design shall be based on 10 CFR 50.44 requirements. Components furnished to satisfy this requirement shall be safety grade.

ABB-CE states in CESSAR-DC Section 20.2.111 that the System 80+ design incorporates a containment building that includes dedicated penetrations for two hydrogen recombiners outside the containment (see CESSAR-DC Section 6.2.5 for the staff evaluation of the containment combustible gas control system). There are two penetrations for each recombiner, one for the line withdrawing combustible gas from the containment and one for the line returning the inerted gas to the containment. These penetrations are designed in accordance with the requirements of GDC 54 and 56.

Specifically, for GDC 54, the lines penetrating the containment have the required isolation and testing capabilities. Each line has two containment isolation valves in series, and test connections allow periodic leak detection tests to be performed.

In accordance with GDC 56, each hydrogen recombiner line has one automatic isolation valve inside the containment (MOV in the line leaving the containment and check valve in the return line) and one motor-operated isolation valve outside the containment. These valves are normally closed with power removed under administrative control (see CESSAR-DC Table 6.2.4-1).

In addition, the penetrations are designed and are of the proper size for the hydrogen recombiner flows as required by 10 CFR 50.44 (see CESSAR-DC Section 6.2.5).

In the DSER, the staff stated that ABB-CE's resolution to this issue for the System 80+ design appeared to be in compliance with 10 CFR 50.34(f). Final determination was contingent upon resolution of all open and confirmatory issues identified in Section 6.2.4 of the DSER on the containment isolation system (CIS). The staff has resolved all open and confirmatory items in Section 6.2.4 of this report and accepted the containment combustible gas control system in Section 6.2.5 of this report; therefore, the staff concludes that System 80+ design complies with 10 CFR 50.34(f).

Therefore, Issue II.E.4.1 is resolved for the System 80+ design.

Issue II.E.4.2: Containment Isolation Dependability

Issue II.E.4.2, in NUREG-0933, addressed improving the reliability and capability of containment structures to reduce the radiological consequences to the public from accidents, including degraded core events. The issue specifically addressed the need for dependable isolation of containment penetrations. The requirements for this issue in NUREG-0737 that are applicable to the System 80+ design follow:

- CIS designs shall have diversity in the parameters sensed for the initiation of containment isolation in accordance with SRP Section 6.2.4, "Containment Isolation System."

- All plant personnel shall identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify the containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

- All nonessential systems shall be automatically isolated by the containment isolation signal.

- The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

- The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

- Containment purge valves that do not satisfy the operability criteria in BTP (CSB) 6-4 of SRP Section 6.2.4 or in the staff interim position of October 23, 1979, must be sealed closed as defined in SRP Section 6.2.4, Issue II.3.f, during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified closed at least every 31 days.

- Containment purge and vent isolation valves must close on a high radiation signal.

In addition, NUREG-0737 added the following requirements to the above 7 positions on the issue:

(1) For postaccident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of GDC 54, 55, 56, and 57, as clarified by SRP Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by SRP Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals.

(2) Revision 2 to RG 1.141, "Containment Isolation Provisions for Fluid Systems," contains guidance on the classification of essential versus nonessential systems.

(3) Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4 above.

(4) Ganged reopening of containment isolation valves is not acceptable. Isolation valves must be reopened on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

(5) The containment pressure history during normal operation should serve as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside the containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 6.9 kPa (1 psi) above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 6.9 kPa (1 psi) will require detailed justification. Applicants for an operating license should use pressure-history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint pressure.

(6) Sealed-closed purge isolation valves shall be under administrative control to ensure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

ABB-CE states in CESSAR-DC Section 20.2.112 that the System 80+ standard design incorporates a CIS for fluid systems piping and for the containment purge ventilation system. ABB-CE stated that I&C sensing lines that penetrate the containment building have containment isolation provisions that meet the intent of RG 1.11, except for such lines as the four pressure instrument sensing lines in the containment building, which are exempt (see CESSAR-DC Section 6.2.4.1.1).

The CIS is designed to prevent or limit the release of radioactivity to the environment during and after an accident while ensuring continued operability of safety-related systems that might be needed to limit or prevent the consequences of an accident. The CIS is, in fact, not a single system but comprises various containment penetrations whose isolation valve arrangement is uniformly designed, fabricated, and tested according to the criteria specified above. A more detailed description of the CIS for the fluid systems piping is in CESSAR-DC Section 6.2.4. In CESSAR-DC Section 9.4.5, ABB-CE describes that part of the CIS which addresses the containment purge ventilation system.

It is the position of ABB-CE that the System 80+ standard design meets the acceptance criteria in the following ways:

- The CIS for the fluid systems piping and containment purge ventilation system ducting meets the intent of SRP Section 6.2.4 (Rev. 2) and the supplemental guidance identified in BTP (CSB) 6-4, including the requirements in GDC 1, 2, and 4 (see CESSAR-DC Sections 3.1.1, 3.1.2, and 3.1.4, respectively).

- With regard to GDC 16, which addresses maintaining the leak-tightness of the containment building, the containment building is designed to protect the public from the consequences of an accident (i.e., minimize the release of radioactivity) and to safely withstand all internal and external environmental conditions that may be reasonably expected to occur during the plant's lifetime (see CESSAR-DC Sections 3.1.12 and 6.2.4, respectively).

- The CIS conforms to the requirements of ANSI N271-1976 and, thus, meets the intent of RG 1.141 and the requirements in GDC 54 through 57, for the isolation of fluid systems. The system's design basis addresses such requirements as leak detection, isolation, and leakage containment capabilities. It also establishes such design features as redundant and reliable isolation valves, and defines the system's performance requirements (see CESSAR-DC Section 6.2.4.2).

- In accordance with NUREG-0737, the design of I&C systems for the automatic containment isolation valves is such that resetting the isolation signal does not result in the automatic reopening of the valves. Reopening of containment isolation valves requires deliberate operator action to open valves on an individual containment penetration basis (see CESSAR-DC Section 6.2.4.5).

- In addition to fluid systems piping, the CIS also includes the containment purge ventilation system that must be isolated during a LOCA. This system is designed to provide a means of purging and venting the containment building whenever the containment is or will be occupied by plant personnel, such as for plant refueling and extended maintenance activities. The system is, therefore, designed to meet the intent of BTP (CSB) 6-4 (which references GDC 54 and 56) with respect to maintaining containment integrity during and after a LOCA (see CESSAR-DC Section 9.4.6).

In DSER Section 6.2.4, the staff stated that ABB-CE had not provided an acceptable resolution of Issue II.E.4.2. In order to satisfactorily resolve this issue, System 80+ had to comply with SRP Section 6.2.4. As discussed in Section 6.2.4 of this report, ABB-CE resolved the outstanding open and confirmatory issues identified in Section 6.2.4 of the DSER. As a result, the requirements in SRP Section 6.2.4 have been met by the System 80+ design.

The staff evaluated and approved the CIS in Section 6.2.4 of this report and, therefore, Issue II.E.4.2 is resolved for the System 80+ design.

Issue II.E.4.4: Purging

Issue II.E.4.4, in NUREG-0933, addressed a reevaluation of the acceptability of purging/venting nuclear power plant containments during the reactor operating modes of startup, power operation, hot standby, and hot shutdown. The three applicable subissues are listed below.

- Issue II.E.4.4(1): <u>Issue Letter to Licensees Requesting Limited Purging</u>

  Issue II.E.4.4(1) addressed the letter issued by NRC to all licensees on November 28, 1978, requiring compliance with specific requests (enclosed in that letter) about containment purging, or venting, during reactor power operation. A number of events had occurred over a span of several years preceding 1979 were directly related to containment purging during normal plant operation. Some of these events raised questions relating to automatic isolation of the purge penetrations that are used during power operation. At Millstone Unit 2, intermittently, the containment was purged with the safety actuation isolation signals to both inboard and outboard containment isolation valves in the purge system inlet and outlet lines manually overridden and inoperable. At Salem Unit 1, the containment was vented through the containment ventilation system valves to reduce pressure. In certain instances, this venting occurred with the containment high particulate radiation monitor isolation signal to the purge and pressure vacuum relief valves overridden.

  These events raised concerns relative to potential failures affecting the purge penetration valves that could lead to a degradation in containment integrity and, for PWRs, a degradation in ECCS performance because of insufficient containment back pressure.

  In the letter of November 28, 1978, NRC requested all licensees of operating reactors to respond to generic concerns about containment purging or venting during normal plant operation. The generic concerns were twofold:

- Events had occurred where licensees overrode or bypassed the safety actuation isolation signals to the containment isolation valves. These events were determined to be abnormal occurrences and reported to Congress in January 1979.

- Recent licensing reviews have required tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a DBLOCA.

In the November 1978 letter, the staff requested that licensees take the following actions pending completion of the NRC review: (1) prohibit the override or bypass of any safety actuation signal which would affect another safety actuation signal and (2) cease purging (or venting) the containment or limit such activity to an absolute minimum, not to exceed 90 hours per year. Licensees were asked to demonstrate (by test or by test and analysis) that containment isolation valves would shut under postulated DBLOCA condition. The NRC positions were amplified by citation of SRP Section 6.2.4 (Rev. 1) and the associated BTP (CSB) 6-4, which have effectively classed the purge and vent valves as "active," invoking the operability assurance program of SRP Section 3.9.3.

- Issue II.E.4.4(2): <u>Issue Letter to Licensees Requesting Information on Isolation Valve</u>

Issue II.E.4.4(2) addressed the letter issued by NRC to all licensees on October 22, 1979, requiring compliance with the staff's interim position (enclosed in that letter) on isolation valve operability for containment purging, or venting, during reactor power operation. After issuing the letter of November 28, 1978, and as a result of site visits, meetings, and telephone conferences with licensees and valve manufacturers, the NRC determined that an interim commitment from all licensees of operating plants was warranted. In the letter of October 22, 1979, the interim NRC staff position was explained as follows, for containment purge and vent valve operation pending resolution of isolation valve operability issues:

- Whenever the containment integrity is required, emphasis should be placed on operating the containment in a passive mode as much as possible and on limiting all purging and venting times to as low as achievable. To justify venting or purging, there must be an established need to improve working conditions to perform a safety-related surveillance or safety-related maintenance procedure. (Examples of improved working conditions include deinerting, reducing temperature[1], reducing humidity[1], and reducing airborne activity sufficiently to permit efficient performance or to significantly reduce occupational radiation exposures), and

- Maintain the containment purge and vent isolation valves closed whenever the reactor is not in the cold-shutdown or refueling mode until such time as the licensee can show that:

  - All isolation valves greater than 7.62-cm (3-in.) nominal diameter used for containment purge and venting operations are operable under the most severe design-basis-accident flow condition loading and can close within the time limit stated in the plant TSs, design criteria, or operating procedures. The operability of butterfly valves may, on an interim basis, be demonstrated by limiting the valve to be no more than 30 degrees to 50 degrees open (90 degrees being fully open). The maximum opening shall be determined in consultation with the valve supplier. The valve opening must be such that the critical valve parts will not be damaged by DBLOCA loads and that the valve will tend to close when the fluid dynamic forces are introduced, and

  - Modifications, as necessary, have been made to segregate the containment ventilation isolation signals to ensure that, as a minimum, at least one of the automatic SIASs is uninhibited and operable to initiate valve closure when any other isolation signal may be blocked, reset, or overridden.

- Issue II.E.4.4(3): <u>Issue Letter to Licensees on Valve Operability</u>

Issue II.E.4.4(3) addressed the letter issued by NRC to all licensees on November 28, 1978 (i.e., the same letter sent for Issue II.E.4.4(1)) requiring compliance with the staff's position (enclosed in that letter) against full opening of isolation valves for containment purging, or venting, during reactor power operation. In the letter of November 28, 1978, the staff asked all licensees of operating reactors to respond to generic concerns about containment purging and venting during normal plant operation. By reviewing responses, NRC learned that at least three valve vendors reported that

[1] Only where temperature and humidity controls are not in the present design.

their valves may not close against ascending differential pressure and the resulting dynamic loading of the DBLOCA. For plants using valves from these vendors, the staff determined that the containment integrity could be sufficiently assured by maintaining the valves in the closed position or by restricting the angular opening of the valves whenever primary containment integrity is required.

NRR sent a letter to all licensees of operating plants on September 27, 1979, requesting compliance with the specific guidelines enclosed with that letter. All licensees that used valves identified by the three manufacturers as having potential closure problems were required to either maintain the valves closed or to install devices to limit the opening angle at all times when containment integrity is required, until such time that full opening was justified to the NRC. The guidelines for demonstrating operability of purge and vent valves were issued as follows:

I. Operability

In order to establish operability it must be shown that the valve actuator's torque capability has sufficient margin to overcome or resist the torques and/or forces (i.e., fluid dynamic, bearing, seating, friction) that resist closure when stroking from the initial open position to full seated (bubble tight) in the time limit specified. This should be predicated on the pressure(s) established in the containment following a DBLOCA. Considerations that should be addressed in assuring valve design adequacy included:

- valve closure rate versus time (i.e., constant rate or other)
- flow direction through valve; ΔP across valve
- single valve closure (inside containment or outside containment valve) or simultaneous closure (establish worst case)
- containment back-pressure effect on closing torque margins of air operated valves that vent pilot air inside containment
- adequacy of accumulator (when used) sizing and initial charge for valve closure requirements
- for valve operators using torque limiting devices whether or not the settings of the devices are compatible with the torques required to operate the valve during the design-basis condition
- effect of the piping system (turns, branches) upstream and downstream of all valve installations
- effect of butterfly valve disc and shaft orientation to the fluid mixture egressing from the containment

II. Demonstration

Various aspects of operability of purge and vent valves may be demonstrated by analysis, bench testing, in situ testing, or a combination of these means.

Purge and vent valve structural elements (valve/actuator assembly) must have sufficient stress margins to withstand loads imposed while valve closes during a DBA. Torsional shear, shear, bending, tension, and compression loads/stresses should be considered. Seismic loading should be addressed.

Once valve closure and structural integrity are assured (by analysis or testing, or by a suitable combination thereof), the sealing integrity after closure and long-term exposure to the containment environment should be evaluated. Emphasis should be directed at the effect of radiation and of the containment spray chemical solutions on seal material. Other aspects, such as the effect on sealing from outside ambient temperatures and debris, should be considered.

III. Bench Testing

The following considerations apply when bench testing was chosen as a means for demonstrating valve operability:

A. Bench testing can be used to demonstrate suitability of the in-service valve by reason of its traceability in design to a test valve. When qualifying valves through bench testing; consider whether or not

   - a valve was qualified by testing of an identical valve assembly or by extrapolation of data from a similarly designed valve.
   - measures were taken to assure that piping upstream and downstream and valve orientation are simulated.
   - the following load and environmental factors were considered

     - simulation of LOCA
     - seismic loading
     - temperature soak
     - radiation exposure

- chemical exposure
- debris

B. Bench testing of installed valves to demonstrate the suitability of the specific valve to perform its required function during the postulated DBA is acceptable. The factors listed in Items III.A.2 (above) and III.A.3 should have been considered when taking this approach.

IV. In Situ Testing

Purge and vent valves may be tested in situ to confirm the suitability of the valve under actual conditions. When performing such tests, the conditions (loading, environment) to which the valve(s) will be subjected during the test should simulate the DBA.

- Resolution of Subissues II.E.4.4(1), (2), and (3)

ABB-CE addresses these Issues II.E.4.4(1), (2), and (3) in CESSAR-DC Section 20.2.113. It states that designing of the containment purge ventilation system in accordance with SRP Section 6.2.4 is sufficient to resolve these issues.

The containment purge ventilation system in the System 80+ standard plant is designed to supply clean, fresh air whenever the containment or incore instrumentation room or both will be occupied. Containment air is exhausted to the environment through the purge filter trains. The system is described in CESSAR-DC Section 9.4.6, and consists of two sub-systems: high-volume purge and low-volume purge.

The containment high-volume purge subsystem is designed to maintain the average containment air temperature between 16 °C and 32°C (60 °F and 90 °F) during inspection, testing, maintenance, and refueling operations, and to limit the release of any contamination to the environment. This subsystem is not used during power operation.

The containment low-volume purge subsystem is designed to provide air circulation and reduce airborne radioactivity for access during normal operation or after reactor shutdown. This subsystem will be used only on an as-needed basis during power operation.

Each containment penetration for the two subsystems has two isolation valves, one on each side of the containment pressure boundary. The containment purge isolation valves maintain primary containment integrity during a postulated LOCA and meet the intent of the requirements in SRP Section 6.2.4.

The containment low- and high-volume purge subsystems for the System 80+ standard plant are designed to be periodically inspected, tested, and maintained (see CESSAR-DC Section 9.4.5.4). Furthermore, in order to ensure system operability during normal and accident conditions, LCOs are specified (see CESSAR-DC Chapter 16). The use of these systems during power operation will also be minimized to reduce the probability of radiation releases to the public environment.

In the DSER, the staff stated that the resolution of Issues II.E.4.4(1), II.E.4.4(2), and II.E.4.4(3) was acceptable subject to the following conditions:

- Tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a DBLOCA (Item II.E.4.4(1)) (DSER Open Item 220.2-25). Subsequently, the staff reviewed Amendment U of the CESSAR-DC Section 6.2.4.3 and determined that the commitment of ABB-CE to demonstrate, by tests and analyses, that the valves will isolate without degrading containment integrity during the dynamic loads associated with the DBLOCA is acceptable. This resolves DSER Open Item 20.2-25.

- Analysis is required to show that valve operability is assured against ascending differential pressure and resulting dynamic loading of the DBLOCA within the guidance provided above (Item II.E.4.4(2)) (DSER Open Item 20.2-26). Subsequently, the staff reviewed Amendment U of the CESSAR-DC Section 6.2.4.3 and determined that the commitment of ABB-CE to analyze the valves to ensure that they will close against ascending differential pressure and the dynamic loading associated with a DBLOCA is acceptable. This resolves DSER Open Item 20.2-26.

- Resolution of the open items in Sections 6.2.3 and 6.2.4 of the DSER. Subsequently, ABB-CE submitted the information needed to resolve the open and confirmatory items in Sections 6.2.3 and 6.2.4 of the DSER. See Sections 6.2.3 and 6.2.4 of this report.

As a result of the resolution of the open and confirmatory items discussed here, the staff concludes that the System 80+ design adequately resolves the issues identified in Issue II.E.4.4.

Therefore, Issue II.E.4.4 is resolved for the System 80+ design.

Issue II.E.6.1: In Situ Testing of Valves

Issue II.E.6.1, in NUREG-0933, addressed the adequacy of the requirements for safety-related valve testing. Valve performance is critical to the successful functioning of a large number of plant safety systems. This issue was divided into the following four parts:

- testing of PIVs under Issue 105 (discussed in Section 20.3 of this chapter)
- check valve operability
- compliance with the thermal-overload protection provisions for MOVs in RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"
- operability verification for MOVs in accordance with GL 89-10

ABB-CE addresses this issue in CESSAR-DC Section 20.2.114 and states the valve testing program for the System 80+ design is discussed in CESSAR-DC Section 3.9.6. ABB-CE addresses the concerns of the item 1 in the resolution of Issue 105 in Section 20.3 of this chapter, and items 2 and 4 above in CESSAR-DC Sections 3.9.6.2.3 (for check valves) and 3.9.6.2.1 (for MOVs).

In the staff's evaluation of the design, qualification, preoperational and inservice testing of check valves and MOVs, the staff finds that the commitments of ABB-CE provide a reasonable assurance for verifying the operabilities for check valves and MOVs and, therefore, are acceptable. The details of the staff's evaluations are in Sections 3.9.6.2.1 and 3.9.6.2.3 of this report. Moreover, as indicated in CESSAR-DC Table 1.8-1 for item 3 above, the System 80+ design will comply with the guidance of RG 1.106 regarding the application of thermal overload protection devices that are integral with the motor starter for electric motors on MOVs.

Therefore, Issue II.E.6.1 is resolved for the System 80+ design.

Issue II.F.1: Additional Accident Monitoring Instrumentation

Issue II.F.1, in NUREG-0933, addressed providing instrumentation to monitor plant variables and systems during and following an accident. The issue addressed the need for plants to include instrumentation to measure, record, and read out in the control room the following containment parameters:

- pressure
- water level
- hydrogen concentration
- high range radiation
- noble gas effluents

The staff clarified Issue II.F.1 in NUREG-0737 and requirements were issued. The radiation and noble gas effluent instrumentation is required to provide for continuous sampling of radioactive iodine and particulates at all potential accident release points, and for onsite capability to analyze and measure these samples. The acceptance criterion is the guidance in RG 1.97.

ABB-CE addressed this issue in CESSAR-DC Sections 7.5.2.5 and 20.2.115, on the PAMI. The staff's acceptance of this design is discussed in Section 7.5.3 of this report.

Therefore, Issue II.F.1 is resolved for the System 80+ design. See also Issue I.D.5(4) in this section.

Issue II.F.2: Identification of and Recovery From Conditions Leading to Inadequate Core Cooling

Issue II.F.2, in NUREG-0933, addressed the need for plants to install improved accident-monitoring instrumentation for detecting the conditions leading to ICC, such as primary coolant saturation, reactor vessel level, and reactor coolant temperature. The acceptance criterion for the resolution of this issue is that a plant shall have accident-monitoring instrumentation that meets the intent of NUREG-0737. In addition, this instrumentation shall conform to the requirements of GDC 13, 19, and 64, and shall implement the guidance in RG 1.97 (Rev. 3) as related to the detection of and recovery from the conditions leading to ICC and provide unambiguous indication of these conditions.

Specifically, the accident-monitoring instrumentation shall be designed so that the operator will get sufficient information during accident situations to take planned manual actions, and to determine whether safety systems are operating properly. In addition, the instrumentation will also provide sufficient data for the operator to be able to evaluate the potential for core uncovery and gross breach of protective barriers, including the resultant release of radioactivity to the environment.

ABB-CE states in CESSAR-DC Section 20.2.116 that the System 80+ design utilizes the Nuplex 80+ control room, which employs an integrated information display hierarchy to present both safety-related and non-safety-related plant data for monitoring and control by the operator. All information is integrated (in accordance with RG 1.97) so that the same instrumentation used for accident monitoring is also used for normal plant operation. If an accident scenario develops, this integration allows the operators to diagnose and monitor the event using instruments with which they are the most familiar. The Nuplex 80+ information systems also include automatic signal validation, through cross-channel data comparison to ensure that the process information displayed to the operator is correct. Multiple diverse systems are utilized to process and display the data to ensure that information processing errors are detected and alarmed. This integrated information display hierarchy comprises the following major elements: discrete indication and alarm system (DIAS), DPS, the component control system (CCS), and operator displays.

The ICC monitoring instrumentation is part of the Nuplex 80+ control room and is designed to meet the intent of the guidance in NUREG-0737. The ICC instrumentation and displays give sufficient information to permit the operator to evaluate the potential for core uncovery and gross breach of protective barriers, including the resultant release of radioactivity to the environment. The ICC instrumentation consists of RTDs, pressurizer pressure sensors, and a RVLMS employing heated junction thermocouples (HJTCs) and CETs. The staff's acceptance of this design is discussed in Section 7.5.2.7 of this report. The staff has accepted the use of HJTCs for level measurements in existing ABB-CE plants. ABB-CE also addressed instrumentation to detect conditions leading to ICC in Section 2.4 of CESSAR-DC Appendix 17.8A, "System 80+ Shutdown Risk Report."

The Nuplex 80+ control room displays both safety-related and non-safety related plant information and includes data used for detecting ICC conditions. The Nuplex 80+ control room is designed in accordance with RG 1.97 (Rev. 3) and NUREG-0737, as previously described. The staff evaluated System 80+ information systems important to safety and ICC monitoring instrumentation in Sections 7.5 and 7.5.2.7, respectively, of this report. The staff concludes that ICC indications satisfy Issue II.F.2. Therefore, Issue II.F.2 is resolved for the System 80+ design is acceptable.

Issue II.F.3: Instrumentation for Monitoring Accident Conditions

Issue II.F.3, in NUREG-0933, addressed the adequacy and availability of instrumentation that monitors plant variables and systems during and following an accident that includes core damage. Before the TMI-2 accident, nuclear power generating stations were equipped with accident-monitoring instrumentation using the guidance identified in RG 1.97 (Rev. 1) and ANSI/ANS Standard 4.5, "Criteria for Accident Monitoring Functions in Light Water Cooled Reactors."

The acceptance criterion for the resolution of this issue is that there shall be instrumentation of sufficient quantity, range, availability, and reliability to permit adequate monitoring of plant variables and systems during and after an accident. Specifically, the instrumentation shall conform to the guidance in RG 1.97 (Rev. 3) and ANSI/ANS Standard 4.5 and should provide sufficient information to the operator for (1) taking planned manual actions to shut the plant down safely; (2) determining whether the reactor trip, engineered-safety-feature systems, and manually initiated safety-related systems are performing their intended safety functions (i.e., reactivity control, core cooling, and maintaining RCS and containment integrity); and (3) determining the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, RCPB, and containment) and determining if a gross breach has occurred.

ABB-CE states in CESSAR-DC Section 20.2.117 that the System 80+ design incorporates the Nuplex 80+ ACC, which includes the PAMI. The PAMI is designed in accordance with the intent of the guidance in RG 1.97 (Rev. 3) and ANSI/ANS Standard 4.5. In Section 7.5.1.1.5 of CESSAR-DC, ABB-CE describes the parameters monitored, the number of sensed channels, sensor ranges, indicated range, location, and associated RG 1.97 category. Examples of plant parameters monitored are RCS pressure, primary safety valve position, primary coolant temperature, containment pressure, and site radiation. The staff has accepted the safety-related process display instrumentation and the ESF system parameter, as described in Section 7.5.2.1, of this report. The staff also reviewed the list of process parameters and their corresponding ranges, as described in Section 7.5.2.5 of this report using the parameters from Table 3 of RG 1.97 as the guideline. The staff finds that the list conforms to the RG 1.97 parameters.

Therefore, Issue II.F.3 is resolved for the System 80+ design.

Issue II.G.1: Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

Issue II.G.1, in NUREG-0933, addressed upgrading the emergency power for the pressurizer relief and block valves, and pressurizer level indicators. In accordance with the requirements in NUREG-0737, the pressurizer equipment must be supplied from an emergency source of power in the event of LOOP.

ABB-CE states in CESSAR-DC Section 20.2.118 that the System 80+ design does not have pressurizer relief and block valves. The safety-grade SDS performs a rapid depressurization of the RCS to enable the operator to feed and bleed the RCS during beyond-design-basis events. Since this system is designated Class 1E, the systems and components, including the pressurizer level indication and SDS valves, are powered from Class 1E power sources. Accordingly, if the facility lost offsite power, an emergency power source would power the SDS valves and the pressurizer level indicators. These indicators are in the Nuplex 80+ ACC. This arrangement is acceptable because it conforms to the intent of NUREG-0737. The use of SDS is further discussed in Section 6.7, 7.5, and 19.11 of this report.

Therefore, Issue I.G.1 is resolved for the System 80+ design.

Issue II.J.3.1: Organization and Staffing to Oversee Design and Construction

Issue II.J.3.1, in NUREG-0933, addressed requiring "license applicants and licensees to improve the oversight of design, construction, and modification activities so that they will gain the critical expertise necessary for the safe operation of the plant."

In CESSAR-DC Section 20.2.119, ABB-CE states that this issue is addressed in CESSAR-DC Section 17.1 on QA activities.

The construction of the reactor plant design is a function of the COL applicant; however, the design of the plant is a function of both ABB-CE and the COL applicant. Therefore, the resolution of this issue for the design of the System 80+ has to be addressed.

The construction organization is not addressed in this report. The organizational structure of the site operator, including staffing, is addressed in Section 13.1 of this report.

The QA program and RAP for the design, procurement, and fabrication of the System 80+ plant are evaluated in Sections 17.1, 17.2, and 17.3 of this report. The staff's conclusion is that the fundamental requirements for an acceptable design-QA program are in place.

The organization for the plant beyond the System 80+ design, the construction of the plant, and the modification of the plant are outside the scope of design certification for the System 80+ design. A part of these concerns involve the organization of the owner/operator which is discussed in Section 13.1 of this report; however, the concerns involving design of the plant outside of the System 80+ design and construction do not involve the organization of the site operation. Therefore, the COL applicant will have the responsibility for addressing these concerns as part of the COL licensing process. This is included in COL Action Item 13.1-1, 17.1-1, and 17.2-1.

Therefore, Issue II.J.3.1 is resolved for the System 80+ design.

Issue II.J.4.1: Revise Deficiency Reporting Requirements

Issue II.J.4.1, in NUREG-0933, addressed assuring that all reportable items are reported promptly to NRC and that the information submitted is complete. The issue was resolved when new requirements were issued by NRC in 10 CFR Part 21 and 10 CFR 50.55(e), on July 31, 1991 (56 FR 36091).

In Section 13.5 of this report, the staff evaluated CESSAR-DC Section 13.5 on plant procedures and the resolution of Issues I.C.1, "Short-term Accident and Procedure Review"; I.C.5, "Feedback of Operating Experience"; and I.C.9, "Long Term Plan for Upgrading Procedures." They are resolved for this design in this section.

The plant procedures for adequately reporting in accordance with 10 CFR Part 21 and 10 CFR 50.55(e) are outside the scope of System 80+ design certification. The COL applicant will have the responsibility for having the proper reporting procedures and addressing this issue as part of the licensing process. This is considered a part of the plant procedures development by the COL applicant discussed in Section 13.5 of this report. This is included in COL Action Item 13.5-2.

In CESSAR-DC Table 20.1-1, ABB-CE stated that Issue II.J.4.1 is not relevant to the System 80+ design because it is an operational issue (i.e., the responsibility of the COL applicant) and not applicable to the design of the reactor plant.

Therefore, Issue II.J.4.1 is resolved for the System 80+ design.

Issue II.K.1(3): Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents

Issue II.K.1(3), in NUREG-0933, addressed requesting that licensees prepare operating procedures for recognizing, preventing, and mitigating void formation in transients and accidents to avoid loss of the core-cooling capability during natural circulation. The staff determined, as stated in NUREG-0933, that this issue was covered by Issue I.C.1.

ABB-CE addressed this issue in CESSAR-DC Section 20.2.120 by referring to Issue I.C.1 which requires that the guidance for the evaluation and development of procedures for transients and accidents be provided. In the review of the resolution of Issue I.C.1 for the System 80+ design in this section, the staff found that the EOGs in CEN-152 contain adequate instructions to address "void formation" during transient and accidents. In the EOGs, operators are instructed to continuously monitor for the presence of voids. Any of the following indications is used to identify void existence:

- letdown flow greater than charging flow
- pressurizer level increases significantly more than expected while the pressurizer spray is in operation
- the reactor-vessel-level-monitoring system (RVLMS), which indicates that voiding is present in the reactor vessel,
- the temperature of the unheated thermocouple in the heated-junction-thermocouple system, which indicates saturated conditions in the reactor vessel head

The instructions for void elimination are:

- Locate letdown or verify letdown is isolated to minimize further inventory loss.
- Stop depressurization to prevent further growth of this void.
- Pressurize and depressurize the RCS within the temperature-pressure limits to condense the void and monitor pressurizer level and the RVLMS for tending of RCS inventory.
- If indications of unacceptable RCS voiding continue, and voiding is suspected to exist in the SG tubes, cool the SG by steaming (or blowdown and feeding) and monitor pressurizer level for tending RCS inventory.
- If indications of unacceptable RCS voiding continue, operate reactor vessel head vent to clear trapped noncondensable gases and monitor for tending of RCS inventory.

The guidelines for treating void formation are in the procedures for dealing with the LOCA, SGTR, excess-steam-demand event, and core and RCS heat-removal control of functional recovery guidelines. The instructions for treating void formation are clear and the information is sufficient for the plant-specific EOPS. This meets Issue I.C.1 of NUREG-0737 as discussed in Issue I.C.1 in this section and Section 13.5 in this report.

Therefore, Issue II.K.1(3) is resolved for the System 80+ design.

Issue II.K.1(4d): Review Operating Procedures and Training To Ensure That Operators Are Instructed Not To Rely on Level Alone in Evaluating Plant Conditions

Issue II.K.1(4d), in NUREG-0933, addressed asking licensees to prepare operating procedures to ensure that operators shall not rely on level indication alone in evaluating plant conditions. As stated in NUREG-0933, the staff determined that this issue was covered by Issues I.A.3.1, I.C.1, and II.F.2; however, ABB-CE addressed this issue in CESSAR-DC Section 20.2.120 by referring only to Issue II.F.2.

Issue I.A.3.1, "Revise Scope and Criteria for Licensing Examinations," was implemented by NRC by a rule change to 10 CFR Part 55, "Operator's Licenses," to require simulator examinations as part of the reactor operator licensing examinations. The staff will impose the requirements of 10 CFR 55.45 on simulators on the COL applicant referencing the system 80+ design; therefore, ABB-CE and the staff does not have to address Issue I.A.3.1 for compliance with 10 CFR 52.47(a)(1)(iv).

The resolution of Issue I.C.1, and the acceptance of the EOGs, for the System 80+ design is discussed in this section.

The resolution of Issue II.F.2 for the System 80+ design is also discussed in this section. The staff finds that multiple indications, including pressurizer water and reactor-vessel water-level indications, are used in the EOGs (ABB-CE's report CEN-152) for event diagnosis and accident mitigation. The EOGs are structured to permit integration with the SPDS and the PAMI. The safety function checks in both optimal and functional recovery guidelines in CEN-152 require operators to check safety function criteria against control board parameters to assess the adequacy of core cooling and the effectiveness of mitigation measure. The features chosen for comparison of these safety function status checks were selected from the list of features identified for inclusion on the SPDS. This permits the machine processing of considerably more plant data in assessing the safety function. The capability of multiple indications is included in CEN-152 for event identification and accident mitigation.

Therefore, Issue II.K.1(4d) is resolved for the System 80+ design.

Issue II.K.1(5): Safety-Related Valve Position Description

Issue II.K.1.5, in NUREG-0933, addressed direct position indication of relief and safety valve position in the control room so that the alarming and indication valve status should be clear and unambiguous and should be evaluated for HFE design considerations. Implementation of a well-engineered bypass and an inoperable status indicating system would provide the operator with timely information on the status of the plant safety systems. This operator aid would help eliminate such operator errors as those resulting from valve misalignment due to maintenance or testing errors. The staff determined, in NUREG-0933, that this issue was covered by Issues I.C.2 and I.C.6.

The requirements of Issues I.C.2 and I.C.6 have been implemented the staff's reviews of reactor plant designs and, therefore, ABB-CE and the staff does not have to address them for compliance with 10 CFR 52.47(a)(1)(iv).

In CESSAR-DC Section 20.2.120, ABB-CE states that Issue II.K.1(5) was resolved for the System 80+ design by the ABB-CE EOGs in CEN-152. The staff accepted the EOGs in the resolution of Issue I.C.1 in this section.

In CESSAR-DC Section 20.2.92 on Issue I.D.3, ABB-CE indicates that the System 80+ control room has dedicated alarms to inform the operators when a valve has opened, providing unambiguous, direct indication of an open or partially open safety or relief valve. This information is acceptable, as discussed in the resolution of Issue I.D.3 in this section.

Therefore, Issue II.K.1.5 is resolved for the System 80+ design.

Issue II.K.1(6): Review Containment Isolation Initiation Design and Procedures

Issue II.K.1(6), in NUREG-0933, addressed ensuring that all lines that do not degrade safety features or core cooling capability are isolated upon automatic initiation of SI. The staff determined, in NUREG-0933, and ABB-CE states, in CESSAR-DC Section 20.2.120, that this issue is covered by Issue II.E.4.2.

The resolution of Issue II.E.4.2 for the System 80+ design is discussed in this section and is acceptable.

Therefore, Issue II.K.1(6) is resolved for the System 80+ design.

Issue II.K.1(9): Review Procedures To Assure That Radioactive Liquids and Gases Are Not Transferred Out of the Containment

Issue II.K.1(9), in NUREG-0933, addressed requiring all operating plant licensees to review their procedures to ensure that radioactive fluids are not transferred out of the containment inadvertently, especially upon ESF reset. All applicable systems and interlocks were required to be listed. The staff determined, in NUREG-0933, that this issue is covered by Issue II.E.4.2, which ABB-CE addresses in CESSAR-DC Section 20.2.112 and Issue I.C.6, which was not addressed by ABB-CE.

ABB-CE states, in CESSAR-DC Section 20.2.120, that this issue is only covered by Issue II.E.4.2.

Issue I.C.6 has been implemented in staff's reviews of reactor plant designs, as documented in NUREG-0933, and it does not have to be addressed by ABB-CE and the staff for compliance with 10 CFR 52.47(a)(1)(iv). The resolution of Issue II.E.4.2 for the System 80+ design is discussed in this section and is acceptable.

Therefore, Issue II.K.1(9) is resolved for the System 80+ design.

Issue II.K.1(10): Review and Modify Procedures for Removing Safety-Related Systems From Service

Issue II.K.1(10), in NUREG-0933, addressed the need to improve procedures for removing and restoring safety-related system to service and knowing the operability status of the system. This issue requires compliance with the requirements of the NRC bulletins related to operability determination and criteria needed to be met before removing safety-related equipment from service.

The staff determined, in NUREG-0933, that Issue II.K.1(10) was covered by Issues I.C.2 and I.C.6; however, both Issues I.C.2 and I.C.6 have been implemented in staff reviews of reactor plant designs and do not have to be addressed for compliance with 10 CFR 52.47(a)(1)(iv).

In CESSAR-DC Section 20.2.120, ABB-CE states that this issue was implemented in the System 80+ design during the development of LCOs and surveillance requirements in the TSs. The staff evaluated the System 80+ TSs in Chapter 16 of this report to ensure that the TSs will preserve the validity of design as described in CESSAR-DC and require that equipment essential to prevent accidents are operable. The staff concludes that the TSs are acceptable.

Procedure development is discussed in Section 13.5 of this report. This is considered outside the scope of the System 80+ design certification and is designated COL Action Item 13.5-1.

Therefore, Issue II.K.1(10) is resolved for the System 80+ design.

Issue II.K.1(13): Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items

Issue II.K.1(13), in NUREG-0933, addressed assuring that operating plants had TSs reflecting the requirements in the bulletins issued by the Commission for the TMI Action Plan. ABB-CE states, in CESSAR-DC Section 20.2.120, that this issue was implemented in the System 80+ TSs.

The System 80+ TSs are evaluated in Chapter 16 of this report. The staff reviewed the TSs against the improved ABB-CE standard TSs which incorporated all the requirements of the bulletins for the TMI Action Plan. Therefore, the approved System 80+ TSs incorporate all the appropriate bulletin requirements from the TMI Action Plan. The incorporation of operating experience in bulletins in the System 80+ design is discussed in Section 20.7 of this report.

Therefore, Issue II.K.1(13) is resolved for the System 80+ design.

Issue II.K.1(14): Review Operating Modes and Procedures To Deal With Significant Amounts of Hydrogen

Issue II.K.1(14), in NUREG-0933, addressed requirements in NUREG-0660 on dealing with significant amounts of hydrogen in containment. The staff determined in NUREG-0933 that this issue was covered by Issues II.B.4, II.B.7, II.E.4.1, and II.F.1.

Issues II.B.4 and II.B.7 have been implemented, as documented in NUREG-033, in staff reviews of reactor plant designs and do not have to be addressed by ABB-CE and the staff for compliance with 10 CFR 52.47(a)(1)(iv). The resolution of the remaining Issues II.E.4.1 and II.F.1 are addressed in the discussions on these issues in this section and are considered resolved for the System 80+ design. In CESSAR-DC Section 20.2.120 on Issue II.K.1(14), ABB-CE also states that the latter two issues cover Issue II.K.1(14).

Therefore, Issue II.K.1(14) is resolved for the System 80+ design.

Issue II.K.1(15): For Facilities with Non-Automatic Auxiliary Feedwater Initiation, Provide Dedicated Operator in Continuous Communication with the Control Room to Operate AFW

Issue II.K.1(15), in NUREG-0933, addressed ensuring that the operating plants had reliable communications to the control room to operate the auxiliary feedwater system. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.12 that this issue was covered by Issue II.E.1.2.

The resolution of Issue II.E.1.2 for the System 80+ design is discussed in this section and is acceptable.

Therefore, Issue II.K.1.(14), is resolved for the System 80+ design.

Issue II.K.1(16): Implemented Procedures That Identify Pressurizer PORV "Open" Indications and That Direct Operator To Close Valve Manually at "Reset" Setpoint

Issue II.K.1(16), in NUREG-0933, addressed requiring procedures that identify pressurizer PORV "open" indications and direct operators to close the valve manually at the "reset" setpoint. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by Issues I.C.1 and II.D.3.

The resolutions of Issues I.C 1 and II.D.3 for the System 80+ design are discussed in this section and are acceptable.

The System 80+ design does not include the pressurizer PORVs, as discussed in Issue 70 in Section 20.2 of this chapter. To provide functions equivalent to those of PORVs (releasing noncondensable gases and mitigating consequences of a beyond-design-basis event), ABB-CE includes in CESSAR-DC Section 6.7 a design of the SDS, which is a manually operated, safety-grade system of pressurizer safety valves. The staff evaluated the SDS in Section 6.7 of this report, and required the SDS to be included in the EOGs for the severe mitigating accident before the staff approved the final design.

The review of updated EOGs, including operation of the SDS, is covered by Issue I.C.1 in this section. The ALMS for safety valve indication is discussed in Issue II.D.3 in this section and Section 7.7 of this report. The ALMS provides the control room operator with direct and unambiguous indication of the position of the pressurizer safety valves.

Therefore, Issue II.K.1(16) is resolved for the System 80+ design.

Issue II.K.1(24): Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip

Issue II.K.1(24), of NUREG-0933, addressed requiring PWR licensees to perform a LOCA analysis for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by Issue I.C.1.

In reviewing the resolution of Issue I.C.1 in this section, the staff finds that the analyses to characterize the effect of RCP operation on SBLOCAs and non-LOCA transients were included in CEN-114, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," and in CEN-115, "Response to NRC IE BL 79-06C, Items 2 and 3, for Combustion Engineering Nuclear Steam Supply Systems." Two analytical models were used in the analyses: a staff-approved ECCS evaluation model was used for analyses under Appendix K requirements to determine compliance with 10 CFR 50.46 acceptance criteria and a best-estimate ECCS model was used to assess the system response.

The analyses primarily addressed the effects of (1) the number of operating RCPs, (2) worst-break location, (3) worst-break size, and (4) high-pressure SI flow rate. These analyses showed that for SBLOCAs, it is beneficial to trip all RCPs in the interest of minimizing the loss of coolant from the primary system. The worst SBLOCA in this regard is a break in the hot leg. ABB-CE used the results of the analyses to support its RCP operating strategy, as discussed in CEN-268, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients." Because CEN-268 is approved by the staff (See the discussion on CEN-128 in Issue II.K.3(5) later in this section) for the adequacy of the RCP operating strategy, the analyses to identify the effect of RCP trip and the trip delay time on SBLOCAs are acceptable.

Therefore, Issue II.K.1(24) is resolved for the System 80+ design. The Issue I.C.1 is discussed in this section and is resolved for the System 80+ design.

Issue II.K.1(25): Develop Operator Action Guidelines Position and Resolution

Issue II.K.1(25), in NUREG-0933, addressed requiring PWR licensees to develop operator action guidelines based on the analyses performed in response to Issue II.K.1(24), which is discussed above. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by Issue I.C.1.

The resolution of Issue I.C.1 for the System 80+ design is in this section and is acceptable. Therefore, Issue II.K.1(25) is also resolved for the design.

Issue II.K.1(26): Revise Emergency Procedures and Train Reactor Operators (ROs) and Senior Reactor Operators (SROs)

Issue II.K.1(26), in NUREG-0933, addressed requiring all operating PWRs to revise their EOPs and to train the ROs and SROs for the plant. The staff determined in

NUREG-0933 that this issue is covered by Issues I.A.3.1, I.C.1, and I.G.1; however, ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by only Issue I.C.1.

The resolution of Issue I.C.1 for the System 80+ design is in this section and is acceptable.

As stated in NUREG-0933, both Issues I.A.3.1 and I.G.1 have been implemented in the staff review of reactor plant designs and do not have to be addressed by ABB-CE for compliance with 10 CFR 52.47(a)(1)(iv).

Issue I.A.3.1 was to revise the scope of examinations and criteria for licensing examinations and Issue I.G.1 was new training requirements for operators. They are the responsibility of the COL applicant and not the responsibility of ABB-CE in the System 80+ design certification. Issue I.A.3.1 is covered by the discussion of training in Section 13.2 of this report and the responsibility of the COL applicant is part of COL Action Item 13.2-1. Issue I.G.1 is discussed in Section 13.2.4 of the FSER and the responsibility of the COL applicant is also part of COL Action Item 13.2-1.

Therefore, Issue II.K.1(26) is resolved for the System 80+ design.

Issue II.K.1(27): Provide Analysis and Develop Guidelines and Procedures for Inadequate Core Cooling

Issue II.K.1(27), in NUREG-0933, addressed requiring PWR licensees to provide analyses and develop guidelines and procedures for an ICC condition. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by Issues I.C.1 and II.F.2.

To satisfy the requirements of Issue II.F.2, ABB-CE describes the ICC instrumentation as consisting of RTDs, pressurizer pressure sensors, CETs, and RVLMS in CESSAR-DC Section 7.5.1.1.7. The signals from the RTDs, pressurizer pressure sensors, and the RVLMS are combined to indicate the subcooling of the reactor coolant. The RVLMS also supplies information to the operator on changes of the liquid inventory in the reactor vessel (RV) regions above or below the fuel alignment plate, and on existence of voiding in the reactor core. The CETs monitor the change of steam temperatures associated with ICC. The resolution of Issue II.F.2 is in this section.

With the ICC system satisfying the Issue II.F.2 requirements, ABB-CE addresses Issue II.K.1(27) by referring to the resolution of Issue I.C.1. In reviewing the resolution to Issue I.C.1, the staff found that the ICC instrumentation is included in CEN-152. These EOGs contain criteria for ECCS termination to ensure RCS inventory conservation without causing overpressurization, and criteria for the RCP restartup to ensure the optimum operating strategy for RCS heat removal and RC inventory control. The guidelines will alert operators to activities that are ineffective or inappropriate for avoiding an ICC situation. The ICC portion of the EOGs clearly describe performance and indicating characteristics of ICC instrumentation, and information regarding the ICC procedures guidelines is sufficient to permit plant-specific operating procedures to be written. The resolution of Issue I.C.1 is in this section.

Therefore, Issue II.K.1(27) is resolved for the System 80+ design.

Issue II.K.1(28): Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required

Issue II.K.1(28), in NUREG-0933, addressed the required design that will ensure automatic RCP trip for all circumstances where required. The staff determined in NUREG-0933 and ABB-CE states in CESSAR-DC Section 20.2.120 that this issue was covered by Issue II.K.3(5).

The resolution of Issue II.K.3(5) for the System 80+ design is later in this section and is acceptable. Therefore, Issue II.K.1(28) is resolved for the System 80+ design.

Issue II.K.3(2): Report on Overall Safety Effect of PORV Isolation System

Issue II.K.3(2), in NUREG-0933, addressed requiring applicants to document the action to be taken to decrease the probability of a SBLOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce challenges to the safety valves for DBE. The requirements were issued in NUREG-0737.

ABB-CE states in CESSAR-DC Section 20.2.121 that the System 80+ design does not include PORVs and, therefore, this issue is not applicable to the System 80+ design. However, to satisfy the staff requirements in SECY-90-016 regarding rapid depressurization for mitigation of beyond-design-basis events, ABB-CE includes the SDS for the System 80+ design.

The SDS is a safety grade system and is manually operated to mitigate beyond-design-basis events. The SDS has two trains, each with two MOVs in series. These valves are manually operated from the control room and the opening of one train would result in a medium LOCA with the plant response being the same as for any other medium LOCA. In Section 6.7 of this report, the staff approved the SDS.

The general ways in which an SDS LOCA might occur is the mechanical failure of the series valves or the inadvertent opening of the valves. As discussed under Issue II.D.3 in this section and Section 7.7 of this report, there is position indication for these valves, so the opening of an SDS train would be indicated to the control room operators. The probability that an SDS valve train will fail open and a medium LOCA will occur is discussed in Section 19.3.3.1 of CESSAR-DC and the staff concluded that the ABB-CE evaluation was acceptable.

Therefore, Issue II.K.3(2) is resolved for the System 80+ design.

For DSER Open Item 20.3-1, in a footnote to the table in DSER Section 20.3, the staff stated that Issue II.K.3(2) would be evaluated in this report. The above discussion addresses this issue and this part of DSER Open Item 20.3-1 is resolved.

Issue II.K.3(5): Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident

Issue II.K.3(5), in NUREG-0933, addressed requiring PWR licensees to study the need for an automatic trip of the RCPs, and to modify procedures or the design, as appropriate. Licensees should know how to operate the RCP in order to mitigate transients and accidents. Preservation of the maximum RCS inventory should be considered in the SBLOCA mitigation; the most effective strategy for DHR should be considered in the other transients' mitigation.

ABB-CE proposes CESSAR-DC Section 20.2.121 that the RCP operating strategy described in report CEN-268 be applied to the System 80+ design. In CEN-268, ABB-CE justifies the use of the trip two/leave two manual RCP trip strategy during transients at ABB-CE plants. The RCP operating strategy is to trip all RCPs in the event of a LOCA and to maintain two RCPs operating during non-LOCA depressurization. The report was approved in GL 86-06, "Implementation of TMI Action Plan Item II.K.3.5, Automatic Trip of reactor Coolant Pumps," dated May 29, 1986.

The Combustion Engineering Owners' Group (CEOG) submitted Revision 1 of CEN-268, and of its Supplement 1, in a letter dated May 21, 1987. Based on its review of Revision 1, the staff concludes that the revisions to CEN-268 do not have any major impact, as far as reactor safety is concerned, on the methodology in CEN-268. Therefore, the staff also concludes that CEN-268 Revision 1, and CEN-268 Supplement 1 Revision 1, are acceptable for implementation of the RCP trip strategy into CEN-152 for EOGs.

Therefore, Issue II.K.3(5) is resolved for the System 80+ design.

Issue II.K.3(6): Instrumentation To Verify Natural Circulation

Issue II.K.3(6), in NUREG-0933, addressed requiring licensees to provide instrumentation to verify natural circulation during transient conditions. The staff determined, in NUREG-0933, that this issue was covered by Issues I.C.1, II.F.2, and II.F.3; however, ABB-CE states in CESSAR-DC Section 20.2.121 that Issue II.K.3(6) was covered by only Issues I.C.1 and II.F.2.

The resolutions of Issues I.C.1, II.F.2, and II.F.3 are in this section and accepted for the System 80+ design.

In reviewing the resolution of Issue I.C.1, the staff finds that the EOGs for natural circulation verification were given in CEN-152 as follows:

If no RCPs are operating, then the operators verify natural circulation flow in <u>all</u> of the following

- loop ΔT (T_h-T_c) less than normal full power ΔT
- hot-leg and cold-leg temperatures constant or decreasing
- RCS subcooling at least 11 °C (20 °F) based on average CET temperature
- no abnormal difference (greater than 6 °C (10 °F)) between T_h RTDs and CET temperature.

If these criteria are not met, the operators are required to control the plant conditions to prevent violation of a safety function by following the guidelines for RCS pressure and inventory control. These guidelines were included in recovery guidelines for events of LOCA, excess steam demand, SGTR, and RCS and core-heat removal control of functional recovery guidelines in CEN-152. The guidelines are clear, and multiple indications are used for natural circulation verification.

In addition, the staff determines in the review of the resolutions to Issues II.F.2 and II.F.3 that adequate instrumentation (including the RTD and CET) is provided for detecting and mitigating ICC conditions.

Therefore, Issue II.K.3(6) is resolved for the System 80+ design.

Issue II.K.3(8): Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs

Issue II.K.3(8), in NUREG-0933, addressed further staff consideration of the need for diverse DHR methods independent of SGs. The staff determined in NUREG-0933 that this issue was covered by Issues II.C.1 ("Interim Reliability Evaluation Program") and II.E.3.3 ("Coordinated Study of Shutdown Heat Removal Requirements"); however, ABB-CE states in CESSAR-DC Section 20.2.121 that Issue II.K.3(8) was covered by Issue A-45.

As stated in NUREG-0933, Issues II.C.1 and II.E.3.3 have been implemented in the staff review of reactor plant designs and do not have to be addressed for compliance with 10 CFR 52.47(a)(1)(iv). In NUREG-0933, the staff also stated that Issue II.E.3.3 was addressed in Issue A-45.

The resolution of Issue A-45 for the System 80+ design is in Section 20.2 of this chapter and is acceptable.

Therefore, Issue II.K.3(8) is resolved for the System 80+ design.

Issue II.K.3(25): Effect of Loss of AC Power on Pump Seal

Issue II.K.3(25) in NUREG-0933, addressed requiring that BWR licensees determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the RCP seal coolers. Adequacy of the seal design to withstand a LOOP should be demonstrated. This position should prevent excessive loss of RCS inventory following an anticipated operational occurrence.

ABB-CE states CESSAR-DC Section 20.2.121 that the RCP seals are normally cooled by redundant systems: seal injection from the CVCS and CCWS. In the event of LOOP, seal injection can be restored by manually aligning Class 1E power to the normal charging pump and CCWS pump, or by using the positive displacement dedicated seal injection pump. Two of the four CCWS pumps can be powered from the EDGS to cool the RCP seals.

During a complete loss of ac power (i.e., loss of both offsite power and the diesel generators), power can be supplied to the dedicated seal injection pump, one charging pump, and one CCWS pump from the onsite ac power source described in CESSAR-DC Section 8.3.1.1.5. ABB-CE states that the use of redundant, diverse seal cooling systems with multiple electrical power sources significantly reduces the probability of losing seal cooling for the RCPs.

The requirements for this issue in NUREG-0737 are that the consequences of a loss of cooling water to the pump seal coolers is determined and the pump seals should be designed to withstand a complete LOOP for at least 2 hours. If seal failure is the consequence of loss of cooling water for 2 hours, an acceptable solution would be providing emergency power to the CCWS pump.

Resolution of this issue also includes the resolution of Issue 23, "Reactor Coolant Pump Seal Failures." The staff reviewed the resolution of Issue 23 in Section 20.2 of this chapter and considers this issue resolved for the System 80+ design.

Therefore, Issue II.K.3(25) is resolved for the System 80+ design.

Issue II.K.3(30): Revise SBLOCA Methods To Show Compliance With 10 CFR Part 50, Appendix K

Issue II.K.3(30), in NUREG-0933, addressed requiring licensees to revise and submit the analytical methods for small-break analysis for compliance with Appendix K to 10 CFR Part 50 for NRC review and approval. The revision should account for comparisons with experimental data, including data from LOFT test and semiscale test facilities. Alternatively, licensees should provide additional justification of the acceptability of present SBLOCA models with LOFT and semiscale test data.

ABB-CE states in CESSAR-DC Section 20.2.121 that report CEN-203, "Response to NRC Action Item II.K.3(30) - Justification of Small-Break LOCA Methods," was developed to demonstrate the continued acceptability of the staff-approved ABB-CE SBLOCA models.

The staff approved CEN-203 in the following NRC letters:

- Dated June 20, 1985, from C. Thomas (NRC) to R. Wells (CEOG), transmitting "Conditional Acceptance for Referencing of Licensing Topical Report CEN-203 (P), Rev. 1"

- Dated February 11, 1987, from D. Crutchfield (NRC) to J.K. Gasper (CEOG), transmitting "Acceptance for Referencing of Licensing Topical Report"

The staff concluded that the currently approved SBLOCA evaluation models are conservative compared with the LOFT and semiscale test data and that they are acceptable for continued use in licensing applications.

Therefore, Issue II.K.3(30) is resolved for the System 80+ design.

Issue II.K.3(31): Plant-Specific Calculations To Show Compliance With 10 CFR 50.46

Issue II.K.3(31), in NUREG-0933, addressed requiring licensees to submit the plant-specific SBLOCA analyses, using the NRC-approved method as described in the preceding Issue II.K.3(30), to show compliance with 10 CFR 50.46.

ABB-CE states in CESSAR-DC Section 20.2.121 that the SBLOCA analysis for the System 80+ design is discussed in CESSAR-DC Section 6.3.3. Report CEN-203, as discussed in Issue II.K.3(30) above, was developed to demonstrate the continued acceptability of ABB-CE's SBLOCA models.

The staff reviewed the SBLOCA analysis and determined, in Section 15.3.7 of this report, that the analysis is acceptable. The staff finds the analysis acceptable because ABB-CE used the NRC-approved evaluation models as described in Issue II.K.3(30) to analyze small breaks and demonstrated that the analytical results comply with the LOCA performance criteria of 10 CFR 50.46, which require that the peak cladding temperature is less than 1204 °C (2200 °F), the maximum local cladding oxidation of 17 percent, and the maximum core-wide oxidation of 1.0 percent of the total amount of the metal in the core.

Therefore, Issue II.K.3(31) is resolved for the System 80+ design.

Issue II.K.3(55): Operator Monitoring of Control Board

Issue II.K.3(55), in NUREG-0933, addressed operator monitoring in the control room for all Westinghouse and Combustion-Engineering plants; however, no requirements were issued in NUREG-0737. The staff determined in NUREG-0933 that this issue was addressed by Issues I.C.1, I.D.2, and I.D.3; however, ABB-CE states in CESSAR-DC Section 20.2.122 that Issue II.K.3(55) is only covered by Issue I.C.1 and I.D.3.

The resolutions of Issues I.C.1, I.D.2, and I.D.3 for the System 80+ design are in this section and are acceptable; therefore, Issue II.K.3(55) is resolved for the System 80+ design.

Issue III.A.1.2: Upgrade Licensee Emergency Support Facilities

Issue III.A.1.2, in NUREG-0933, addressed requiring licensees to upgrade their emergency support facilities by establishing a technical support center (TSC), an operational support center (OSC), and a nearsite emergency operations facility (EOF) for command and control, support, and coordination of onsite and offsite functions during reactor accident situations.

ABB-CE discusses only the TSC in CESSAR-DC Section 20.2.122 in addressing Issue II.A.1.2; however, as discussed in Section 13.3 of this report, the System 80+ design provides for a TSC and an OSC. The staff considers that the nearsite EOF is outside the scope of the System 80+ design certification and will be addressed by the COL applicant, This is COL Action Item 13.3-2.

Therefore, Issue III.A.1.2 is resolved for the System 80+ design.

Issue III.A.3.3: Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Radio Communication Systems

Issue III.A.3.3, in NUREG-0933, addressed upgrading the communications capability at the emergency support facilities at the plant listed in Issue III.A.1.2. This capability is outside the scope of the System 80+ design certification and will be addressed by the COL applicant. This is briefly discussed in Sections 9.5.2 and 13.3 of this report. The responsibility of the COL applicant is part of COL Action Item 9.5.2-1.

Therefore, Issue III.A.3.3 is resolved for the System 80+ design.

Issue III.D.1.1: Primary Coolant Sources Outside the Containment Structure

Issue III.D.1.1, in NUREG-0933, addressed identifying design features to reduce the potential for exposure to workers at plants and to offsite populations from the release of primary coolant following an accident. This issue has three subissues:

- III.D.1.1(1), "Review Information Submitted by Licensees Pertaining to Reducing Leakage From Operating Plants"
- III.D.1.1(2), "Review Information on Provisions for Leak Detection"
- III.D.1.1(3), "Develop Proposed System Acceptance Criteria"

The requirements for the first subissue are in NUREG-0737. The other two were concluded to be resolved in NUREG-0933 and were then dropped from further consideration.

The staff determined that the safety concerns raised in Subissue III.D.1.1(2) were addressed in other issues, including Issues 66 and 119.5 which are addressed, and accepted, for the System 80+ design in Section 20.3 in this chapter. The need for requiring leak-detection systems and the development of new acceptance criteria for these systems in Subissue III.D.1.1(3) were pursued by the staff in other issues, as Subissue III.D.1.1(2). Therefore, work on Subissue III.D.1.1(3) did not provide any data for staff consideration and this issue was dropped from further consideration.

In CESSAR-DC Section 20.2.125, ABB-CE considered only Subissue II.D.1.1(2) as relevant to the System 80+ design. It stated that the subissue was resolved by the monitoring provisions summarized in Section 11.5, the ALARA evaluation in Section 12.1, and the radiation protection design features in Section 12.3, of CESSAR-DC. These sections are discussed below.

In NUREG-0737, Subissue III.D.1.1(1) required licensees to implement a program to reduce leakage from systems outside the containment that would or could contain highly radioactive fluids during a serious transient, or following an accident, to as-low-as-practical levels. System 80+ design features are discussed in Sections 12.1 and 12.3 on radiation protection and ALARA (as low as is reasonably achievable), Section 5.2.5.1 on the RCPB leakage detection methods, Section 11.2 on the liquid radwaste management system, and Section 11.5 on process and effluent monitors, of the CESSAR-DC.

For Subissue III.D.1.1(2), the staff also stated in NUREG-0933 that Issue II.F.1 addressed accident monitoring instrumentation and that the RCPB leak detection capability must be equivalent to that specified in RG 1.45. ABB-CE discusses RCPB leakage detection systems in CESSAR-DC Section 5.2.5.1. The staff evaluated these systems in Section 5.2.2 of this report and found them acceptable with respect to RG 1.45. Also, Issue II.F.1 is addressed and resolved for the System 80+ design in this section.

Therefore, Issue III.D.1.1 is resolved for the System 80+ design.

For DSER Open Item 20.3-1, in a footnote to the table in DSER Section 20.3, the staff stated in part that Issue III.D.1.1 would be evaluated in this report. The above discussion addresses this issue and this part of DSER Open Item 20.3-1 is resolved.

Issue III.D.3.3: In-Plant Radiation Monitoring

Issue III.D.3.3, in NUREG-0933, addressed improving radiation protection for nuclear power plant workers. This issue required licensees to improve in-plant radioiodine instrumentation under accident conditions. This issue has the following four subissues:

- III.D.3.3(1), "Issue Letter Requiring Improved Radiation Sampling Instrumentation"
- III.D.3.3(2), "Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment"
- III.D.3.3(3), "Issue Rule Change Providing Acceptable Methods for Calibration of Radiation Monitoring Instruments"
- III.D.3.3(4), "Issue a Regulatory Guide."

For Subissue III.D.3.3(1), the staff stated in NUREG-0737 that licensees shall provide equipment, and associated training and procedures for accurately determining airborne radioiodine concentrations in areas where plant personnel may be during an accident.

For Subissue III.D.3.3(2), the staff stated in NUREG-0933 that the subissue was resolved with the requirements for high-range area and portable monitors being incorporated into Revision 2 of RG 1.97 and SRP Sections 12.3 and 12.5 being revised in July 1981, to incorporate new requirements for in-plant radiation monitoring.

For Subissues III.D.3.3(3) and (4), the staff stated in NUREG-0933 that (1) the revision to 10 CFR 20.501(c) on acceptable methods to calibrate radiation monitoring instruments and (2) RG 8.25, "Air Sampling in the Workplace," being issued in August 1980, resolved these sub-issues, respectively.

ABB-CE states in CESSAR-DC Section 20.2.126 that the permanently installed radiation monitoring system (RMS) of the System 80+ design is described in CESSAR-DC Section 11.5 (with monitor types, sensitivities, ranges, and other data in CESSAR-DC Tables 11.5-1 through -5) and airborne radiation monitors are described in CESSAR-DC Section 11.5.1.2.4.

These monitors include portable units that can be moved to areas where work or surveillance activities are at an unusual risk of airborne exposure. All equipment is assembled on a mobile cart, and the design allows for transfer of sample filters or cartridges to the station counting room for further analysis. The equipment for continuous sampling during and after an accident of plant gaseous effluent for noble gas, radioiodine, and particulates and for radiation monitoring of areas requiring post-accident access is described in the following sections of CESSAR-DC: 11.5.1.2.1, 11.5.1.2.3, 11.5.1.2.4, 11.5.1.2.5, and 11.5.1.2.6.

The staff discusses the area radiation and airborne radioactivity monitoring instrumentation in Section 12.3.4 of this report. In this section, the staff states that the System 80+ design will have portable airborne monitors available to provide accurate determination of airborne radioiodine concentrations in areas that would not be covered by fixed instrumentation. These monitors will meet the equipment requirements discussed above for this issue. The COL applicant will provide the additional information concerning the specific equipment to be used, and the training and procedures that will be followed. This is COL Action Item 12.3.4-2.

Therefore, Issue III.D.3.3 is resolved for the System 80+ design.

Issue III.D.3.4: Control Room Habitability

Issue III.D.3.4, in NUREG-0933, addressed upgrading the habitability of the control room for the operators. The requirements were given in NUREG-0737.

ABB-CE states in CESSAR-DC Section 20.2.127 that the System 80+ control room habitability design is discussed in CESSAR-DC Section 6.4. The staff accepts the design in Section 6.4 of this report and states that the COL applicant will have to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shutdown under conditions created by any DBA. This is outside the scope of the System 80+ design certification and is designated COL Action Item 6.2-4. See also Issue 83 in Section 20.3 of this chapter.

Therefore, Issue III.D.3.4 is resolved for the System 80+ design.

20.5 Human Factors Issues

The resolution of the human factors issues, in NUREG-0933, for the System 80+ design are discussed in detail in Section 18.3.3.2.5 of this report and are mentioned briefly below. These human factors issues were taken from NUREG-0985, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," (Revision 2) dated April 1986. In Chapter 18 of this report, the staff presents its evaluation of the HFE for the System 80+ design.

In the DSER, the staff stated that the human factors issues discussed below would be addressed in this report and designated the action as DSER Action Item 20.2-29. On the basis of the evaluations below, DSER Action Item 20.2-29 is resolved.

Issue HF1.1: Shift Staffing

This issue addressed ensuring that the numbers and capabilities of the staff at nuclear power plants are adequate to operate the plant safely. This issue was to determine the minimum appropriate shift crew staffing composition. To meet this goal, consideration was given to

- the number and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode;
- the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty; and
- appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation.

The review criteria for this issue are contained in the 10 CFR 50.54, SRP Sections 13.1.2 through 13.1.3, and RG 1.114, "Guidance to Operators at the Controls and to

Senior Operators in the Control Room of a Nuclear Power Unit."

ABB-CE states in CESSAR-DC Section Table 20.1-1 that this issue is not relevant to the System 80+ design because NRC had identified this as an operational issue, which is the responsibility of the COL applicant.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design. The resolution involves a responsibility for the COL applicant to adhere to RG 1.114, which is part of COL Action Item 13.1-1.

Issue HF4.4: Guidelines for Upgrading Other Procedures

This issue addressed ensuring that plant procedures are adequate and could be used effectively, and to guide operators in maintaining plants in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing, and criteria for evaluating, EOPs, normal operating procedures, and other procedures that affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

The review criteria for this issue are in SRP Sections 13.5.1 and 13.5.2, and in IN 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures."

ABB-CE states in CESSAR-DC Section Table 20.1-1 that this issue is not relevant to the System 80+ design because NRC had identified this as an operational issue, which is the responsibility of the COL applicant.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design. The resolution of the issue is through procedure development which is discussed in Section 13.5 of this report, and is outside the scope of the System 80+ design certification and the responsibility of the COL applicant (COL Action Item 13.5-1).

Issue HF4.5: Application of Automation and Artificial Intelligence (AI)

This issue concerned the level of automation possible in power plants within the nuclear industry. The level of automation spans a range of possibilities from the fully manual, with locally-operated valves, to the fully-automated, employee artificial intelligence. Reducing the menial level of workload of operators could provide better low-level control and fewer operator errors. Such automation can also free operators to concentrate on the cognitive level of operations. Automation and AI affects the following: control room design, operating procedures, and other operator aids, staffing, and training.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design.

Issue HF5.1: Man-Machine Interface - Local Control Stations

This issue addressed ensuring that the man-machine interface is adequate for the safe operation and maintenance of a nuclear power plant. The concerns associated with this issue include the assurance that indications and controls made available to operators at local control stations outside of the control room or remote shutdown room are sufficient and appropriate for their intended use.

The regulatory guidance has been limited to the control room and the remote shutdown panel. Control room crew activities should be analyzed to establish and describe communication and control links between the control room and the auxiliary control stations. Additionally, the potential impact of auxiliary personnel on plant safety should be analyzed.

ABB-CE states in CESSAR-DC Section 20.2.86 that the design philosophy of the local control stations in Nuplex 80+ are described in CESSAR-DC Section 18.7.1.6.2. Communications between the local stations and the main control room are discussed in CESSAR-DC Section 9.5.2.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design.

Issue HF5.2: Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation

This issue concerned the use of advanced I&Cs, in particular with respect to plant annunciators.

ABB-CE states in CESSAR-DC Section 20.2.87 that the design of the annunciator system is in CESSAR-DC Sections 18.7.1.1.4 and 18.7.1.5. Of major importance is the reduction in the stimulus overload to the operators which can occur in during major transients. They are functionally grouped and prioritized.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design.

Issue HF5.3: Man-Machine Interface - Evaluation of Operational Aids

This issue concerned the development and implementation of additional operator aids into control rooms. The goal is to improve operator performance through the implementation of effective techniques for display of information. The issue encompasses areas such as alarm system enhancements and potential use of advanced computer support techniques such as AI and expert systems.

ABB-CE states in CESSAR-DC Section Table 20.1-1 that this issue is not relevant to the System 80+ design because NRC had identified this as an operational issue, which is the responsibility of the COL applicant.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design; however, the staff does not consider that there is any action needed to be taken by the COL applicant.

Issue HF5.4: Man-Machine Interface - Computers and Computer Displays

This issue concerned the integration of computers and computer-driven displays into new control room designs.

ABB-CE states in CESSAR-DC Section Table 20.1-1 that this issue is not relevant to the System 80+ design because NRC had identified this as an operational issue, which is the responsibility of the COL applicant.

See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is resolved for the System 80+ design; however, the staff does not consider that there is any action needed to be taken by the COL applicant.

20.6 Additional Three Mile Island Action Plan Requirements

Pursuant to 10 CFR 52.47(a)(ii), an applicant for design certification must demonstrate compliance with any technically relevant TMI Action Plan items addressed in 10 CFR 50.34(f). The relevant TMI Action Plan items and the section where they are addressed are listed in Table 20.2 of this report.

The DSER COL Action Item 20.3-1, a footnote to Table 20.2 in the DSER, stated that Issue III.J.3.1 would be addressed by the applicant for COL (i.e., the COL applicant). Issue II.J.3.1 is now addressed in Section 20.3 of this report and it references COL Action Items 13.1-1, 17.1-1, and 17.2-1. Therefore, COL Action Item 20.3-1 is not needed and is closed out.

20.7 Incorporation of Operating Experience

Background

The NRC staff issues generic communications (bulletins, generic letters, and information notices) to transmit operational experience information to industry. A bulletin or generic letter is typically issued when the NRC staff determines that licensees should be required to inform the NRC what actions have been or will be taken to address an event, condition, or circumstance that is both potentially safety significant and generic. An information notice is typically issued when the NRC staff determines that licensees should be informed of an event, condition, or circumstance that may be both potentially safety significant and generic, but the event, condition, or circumstance is not sufficiently significant to warrant requiring licensees to confirm in writing that actions have been or will be taken. Potential safety issues highlighted in NRC generic communications have resulted in the establishment of a USI or GSI, and have also been incorporated into formal regulatory requirements.

Application Content Review

ABB-CE states that it considered operational experience information in the design of the System 80+. In CESSAR-DC Section 1.8, ABB-CE presents the findings of its review of bulletins and generic letters. ABB-CE determined the applicability of the generic letters and bulletins to the System 80+ design, and gave the basis for this determination in CESSAR-DC Tables 1.8-2 and 1.8-3, respectively. As shown in these tables, ABB-CE reviewed the generic letters and bulletins that were issued on or after January 1980. This is acceptable to the NRC staff as discussed later in this section, under the heading, "Regulatory Review." These tables resolve DSER Confirmatory Item 20.4-1.

ABB-CE also states in CESSAR-DC Section 1.2 that information about operational experience obtained from sources other than bulletins and generic letters was incorporated into the System 80+ design. In CESSAR-DC

Table 20.2 52.47(a)(1)(ii) TMI Action Plan Items

TMI REQUIREMENT 50.34(f)	FSER CHAPTER	FSER SECTION
II.B.8	19, 20	19.1, 19.2, 19.4, 20.4
II.E.1.1	10, 19, 20	10.4.9, 19.1, 20.4
II.K.3(2)	20	20.4
I.A.4.2	20	20.4
I.C.9	20	20.4
I.D.1	18, 20	All, 20.4
I.D.2	18, 20	18.7, 20.4
I.D.3	20	20.4
II.B.1	20	20.4
II.B.2	12, 20	12.2.3, 12.3.1, 20.4
II.B.3	20	20.4
II.D.1	20	20.4
II.D.3	20	20.4
II.E.1.2	20	20.4
II.E.3.1	8, 20	8.4, 20.4
II.E.4.2	20	20.4
II.E.4.4	20	20.4
II.F.1	7, 20	7.5, 12.3.4, 20.4
II.F.2	20	20.4
II.F.3	20	20.4
II.G.1	20	8.4, 20.4
III.A.1.2	13, 20	13.3, 20.4
III.D.1.1	20	20.4
III.D.3.3	12, 20	12.3.4, 20.4
I.F.1	17, 20	17.1, 20.4
I.F.2	17, 20	17.1, 20.4
II.E.4.1	20	20.4
II.J.3.1	20	20.4
III.D.3.4	6, 20	6.4, 20.4

Table 1.2-1, ABB-CE describes the incorporation into the System 80+ design of collective industry experience as promulgated through the EPRI Utility Requirements Document, as well as designer-specific experience. This table resolves DSER Confirmatory Item 20.4-2.

Regulatory Review

The SRP (NUREG-0800) guides the NRC staff for its review of a reactor facility design. This document states requirements, acceptance criteria (some of which are based on operating reactor experience), and findings that the staff must make. This document was last revised in April 1982. Significant issues raised before January 1981 were incorporated into the April 1982 revision. Accordingly, the staff concludes that it is appropriate to focus its review on issues of operating experience identified by NRC since January 1981. However, ABB-CE have reviewed and reported on the bulletins and generic letters issued by the NRC on and after January 1980, as to their applicability to the System 80+ design.

As stated above, the bulletins and generic letters address the issues that are of sufficient safety significance to warrant requiring licensees to inform the NRC of the actions they have taken or will take, whereas information notices do not require a response. Accordingly, the NRC staff concluded that it is appropriate to focus its review on bulletins and generic letters.

The staff reviewed the bulletins and generic letters issued since 1980 for incorporation into the staff's review of the System 80+ design. Upon initial review, certain bulletins and generic letters were excluded from the review because they were not relevant to the design of the System 80+ plant, or because they were associated with TMI Action Plan items, USIs or GSIs, or existing rules and regulations and, thus, were already an integral part of the staff's System 80+ design review process. See the resolution of the technically relevant generic issues in NUREG-0933 (i.e., TMI Action Plan items, USIs, and GSIs) for the System 80+ design in Sections 20.2 through 20.4 of this chapter. As examples, BL 80-01, "Operability of ADS Valve Pneumatic Supply," applies only to BWRs; GL 86-14, "Operator Licensing Examinations," relates to operator licensing exam schedules which are the responsible of the owner/operator; GL 86-10, "Implementation of Fire Protection Requirements," is associated with 10 CFR 50.48 and 10 CFR Part 50, Appendix R; GL 89-06, "Task Action Item I.D.2 — Safety Parameter Display System," is associated with a TMI Action Plan item; and GL 84-15, "Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability," is associated with a USI/GSI. There are additional generic letters which transmitted previously issued bulletins and, therefore, were considered duplicates of the bulletins.

The remaining 75 bulletins and generic letters were reviewed to assure that the issues identified had, if appropriate, been incorporated into the staff's System 80+ design review. Where necessary, additional information was sought from ABB-CE. The identified issues were categorized as: (1) not applicable to the System 80+; (2) applicability to the System 80+ still being determined; (3) not a design issue; or (4) applicable to the System 80+ and addressed in CESSAR-DC, in this report, or in both. The resolution of the issues identified in the 75 bulletins and generic letters is summarized in Tables 20.3 and 20.4, respectively, which follows.

Of the 75 bulletins and generic letters, 27 issues are being resolved during the ongoing preparation of TS (see Chapter 16 of this report); 9 issues were determined not applicable to the System 80+, and 39 issues were either not design issues or were already appropriately considered in the System 80+ design.

The staff stated in the DSER that it was still evaluating BL 80-03, "Loss of Charcoal From Standard Type II, 2-inch, Tray Absorber Cells," BL 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to Environment"; BL 80-24, "Prevention of Damage Due to Water Leakage Inside Containment (October 17, 1980 Indian Point Event)"; and GL 81-38, "Storage of Low Level Radioactive Wastes at Power Reactor Sites." Resolution of these issues was identified as DSER Open Item 20.4-1. The staff has completed its review of these bulletins and generic letters and the results are in Tables 20.3 and 20.4, respectively. The staff concluded that the System 80+ design adequately addressed the concerns. Therefore, DSER Open Item 20.4-1 is resolved.

Conclusion

Of the bulletins and generic letters issued on or after January 1980, 75 were identified for possible incorporation into the staff's System 80+ design review. On the basis of its review of these 75 bulletins and generic letters, the NRC staff concludes that operational experience information has been adequately incorporated into the System 80+ design.

Table 20.3 Resolution of bulletins (BLs) issued since 1980 and their applicability to the System 80+ design

Bulletin No. and Title	Staff Resolution
BL-80-03, Loss of charcoal from Standard Type II, 2-inch, tray absorber cells	This issue is resolved for the System 80+ design and is addressed in Section 6.4 of this report. The charcoal tray and screen of the control room emergency zone of the control complex system is made of all-welded construction to preclude the potential loss from the adsorber cells. All ducts and equipment housings are all welded construction and flanged connections will be pressure tight and periodically visually examined and tested. This concern is resolved by the air filtration systems being designed to the criteria in RGs 1.52 and 1.140, as discussed for the System 80+ design in CESSAR-DC Tables 9.4.5 and 9.4.6, and found acceptable by the staff in Section 9.4 of this report.
BL-80-04, Analysis of a pressurized-water reactor (PWR) main steamline break with continued feedwater addition	This issue is resolved for the System 80+ design and is addressed in Sections 6.2 and 15.3.1 of this report. The resolution of Issue 125.II.7 in Section 20.2 of this chapter is related to this issue.
BL-80-05, Vacuum condition resulting in damage to chemical volume control system (CVCS) holdup tanks	This issue is resolved for the System 80+ design and is addressed in Section 11.2 of this report.
BL-80-06, Engineered safety feature (ESF) reset controls	This issue is resolved for the System 80+ design because the design does not utilize an automatic ESF reset and, following initiation, each ESF system must be manually reset.
BL-80-08, Examination of containment liner penetration welds	This issue is no longer a concern. Since the BL issuance, ultrasonic (UT) examination techniques have been substantially improved and the original concern in the BL that UT examination is not as good as radiography is no longer valid.
BL-80-10, Contamination of nonradioactive system and resulting potential for unmonitored, uncontrolled release to environment	This issue is resolved for the System 80+ design. The staff reviewed CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5, and 12.3, and concluded that these sections satisfactorily address the concerns raised in this BL. The System 80+ design features are adequate to (1) detect the contamination of non-radioactive systems and (2) prevent unmonitored and uncontrolled release of radioactive material to the environment. See Sections 11.2, 11.3, and 11.5 of this report.
BL-80-11, Masonry wall design	This issue is resolved for the System 80+ design because seismic Category I masonry walls are not used in the design, as stated in CESSAR-DC Appendix 3.8A, Section 6.2.1.1.

Bulletin No. and Title	Staff Resolution
BL-80-15, Possible loss of emergency notification system with loss of offsite power	This issue is resolved for the System 80+ design. GL-91-14 referred to BL-80-15 and asked licensees to guarantee power to equipment in accordance with BL-80-15 requirements. ABB-CE stated that GL-91-14 included no design requirements for the System 80+ design (Ref. ABB-CE's letter of February 18, 1992, listed in Appendix A of this report). See the resolution of Issue III.A.3.3 in Section 20.3 of this chapter.
BL-80-18, Maintenance of adequate minimum flow through centrifugal charging pumps following secondary-side, high-energy-line rupture	This issue is resolved for the System 80+ design. As discussed in Section 9.3.4 of this report, the CVCS (including charging pumps) is not a safety system in the System 80+ design and no credit is taken for the CVCS in the safety analyses. However, the CVCS does affect reactor coolant pump (RCP) seal cooling which is addressed in Issue 23 in Section 20.2 of this chapter.
BL-80-19, Failures of mercury-wetted matrix relays in reactor protective systems (RPSs) of operating nuclear plants designed by ABB-CE.	This issue is resolved for the System 80+ design because the design uses digital-based instead of relay-based logic in the RPS.
BL-80-20, Failures of Westinghouse Type W-2 spring return to neutral control switches	This issue is resolved for the System 80+ design because the selection of switch types is not a design issue.
BL-80-24, Prevention of damage due to water leakage inside containment	This issue is resolved for the System 80+ design. The System 80+ design has no open-cycle cooling water systems inside containment and the reactor coolant pressure boundary leakage detection systems, evaluated in Section 5.2.5 of this report, have diverse means of monitoring both identified and unidentified leakage inside primary containment. The systems also function to isolate primary containment on certain indications of gross leakage. Procedures and personnel training related to the operation and maintenance of these systems are provided by the COL applicant, and not by ABB-CE, because they are beyond the scope of the design review.
BL-81-01, Surveillance of mechanical snubbers	This issue was resolved by ABB-CE's commitment to ASME Code, Section XI in CESSAR-DC Section 1.8 and Table 1.8-6. Section XI, Article IWF 5300 references ASME/ANSI OM-1987, Part 4 for inservice testing requirements for snubbers. OM-Part 4 was endorsed by the staff via 10 CFR 50.55a(b)(2)(viii).
BL-81-02, Failure of gate-type valves to close against differential pressure	This issue is resolved for the System 80+ design. It is addressed during the staff review of GL-89-10 and discussed in Section 3.9.6 of this report.

Bulletin No. and Title	Staff Resolution
BL-82-02, Degradation of threaded fasteners in the reactor coolant pressure boundary of PWR plants	This issue of the use of molybdenum disulfide lubricant is resolved for the System 80+ design. As stated in Sections 5.2.3 and 5.3.1 of this report, the use of lubricants within the reactor coolant pressure boundary is limited to small amounts for bolting on the reactor and reactor coolant pump internals and as lubrication for closure studs. These lubricants are either graphite or nickel or both in either an alcohol, silicon, or petroleum oil base with tightly controlled limits on halogen and sulfur. In Section 5.2.3, the staff further states that molybdenum disulfide lubricants are not used within the reactor coolant pressure boundary. (Refs. CESSAR-DC Sections 5.2.3 and 5.3.1).
BL-83-03, Check valve failures in raw water cooling systems of diesel generators	This issue is resolved for the System 80+ design and is addressed in Section 3.9.6 of this report.
BL-86-01, Minimum flow logic problems that could disable residual heat removal (RHR) pumps	This issue is resolved for the System 80+ design because minimum flow lines are always available and there are no isolation valves. BL-86-01 addresses the loss of RHR pumps due to single failure of the isolation valve in the mini-flow line. CESSAR-DC Figure 6.3.2-1A shows one isolation valve in each of the shutdown cooling system (SCS) mini-flow line. FSER Section 5.4.3.1 states that "two separate trains for each unit provide redundancy in the SCS." . . . No single active failure to the SCS can prevent at least one complete train of the SCS from being brought on line from control room during normal plant cooldown, a transient or accident, and adequately addressed the concern of the bulletin.
BL-86-03, Potential failure of multiple emergency core cooling system (ECCS) pumps due to single failure of air-operated valve in minimum-flow recirculation line	This issue is resolved for the System 80+ design and addressed in Section 5.4.3 of this report.
BL-88-04, Potential safety-related pump loss	This issue is resolved for the System 80+ design and addressed in Sections 5.4.3 and 6.3.3 of this report.
BL-88-08, Thermal stresses in piping connected to reactor cooling systems	This issue is resolved for the System 80+ design and addressed in Sections 3.9.3.1 and 3.12.5.9 of this report.
BL-88-11, Pressurizer surge line thermal stratification	This issue is resolved for the System 80+ design and addressed in Sections 3.9.3.1 and 3.12.5.10 of this report.
BL-89-03, Potential loss of required shutdown margin during refueling operations	This issue is resolved for the System 80+ design. It is related to refueling operations and is not a design issue.

Bulletin No. and Title	Staff Resolution
BL-90-01, Loss of fill-oil in transmitters manufactured by Rosemount	This issue is resolved for the System 80+ design, which has on-line monitoring capability. This is an effective method to address the loss of fill-oil in the Rosemount transmitter issue.
BL-92-01, Failure of Thermo-Lag 330 fire-barrier system to maintain cabling in wide cable trays and small conduits free from fire damage	This issue is resolved for the System 80+ design. As stated in Section 9.5.1.2.2 of this report, on passive fire-protection features, there will be 3-hour rated fire barriers designed to the acceptance criteria of American Society for Testing and Materials (ASTM) E-119, "Fire Tests of Building and Construction Materials."
BL-92-01, Supplement 1, Failure of Thermo-Lag fire barrier system to perform its specified fire endurance function	See the resolution for BL-92-01 above.
BL-93-02, Debris plugging of emergency core cooling suction strainers	This issue is resolved for the System 80+ design. See the resolution of Issue A-43 on sump suction strainers in Section 20.1 of this chapter.
BL-93-02, Supplement 1, Debris plugging of emergency core cooling suction strainers	See the resolution for BL-93-02 above.

Table 20.4 Resolution of generic letters (GLs) issued since 1980 and their applicability to the System 80+ design

Generic Letter No. and Title	Staff Disposition
GL-80-001, Report on ECCS cladding models	This is resolved for the System 80+ design. Clad swelling models (as described in NUREG-0630) have been incorporated into ABB-CE's evaluation models (Topical Report CEN-132) used to evaluate the System 80+ design. These models have been reviewed and found acceptable by the staff.
GL-80-019, Resolution of enhanced fission gas release concern	This is resolved for the System 80+ design. Fission gas release models are based on ABB-CE's Topical Report CEN-161(b), and have been reviewed and found acceptable by the staff.
GL-80-030, Clarification of the term "operable" as it applies to single-failure criterion for safety systems required by TS	By adoption of the improved ABB-CE Standard TSs (STS), ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-80-035, Effect of a dc power supply failure on ECCS performances	This issue is resolved for the System 80+ design and addressed in Sections 8.3.2 and 15.3.6 of this report.
GL-80-099, TS revisions for snubber surveillance	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-80-106, Report on ECCS cladding models, NUREG-0630	See the resolution for GL-80-001 above.
GL-80-109, Guidelines for SEP soil-structure interaction reviews	This is resolved for the System 80+ design. Concerns from this GL have been incorporated into SRP Section 3.7.2 which was used for the soil interaction review. This is discussed in Section 3.7.2 of this report.
GL-81-38, Storage of low-level radioactive wastes at power reactor sites	This issue is resolved for the System 80+ design. In Section 11.4 of this report the staff concluded that the onsite storage provided in the System 80+ design was acceptable; however, the staff recognizes in GL-81-38 that there may be a need for additional storage onsite storage for low-level radioactive wastes beyond what has been provided for any reactor plant design, including the System 80+ design. This is a site-specific issue because it will depend upon available offsite storage space for low-level radioactive waste from the plant. This will be identified by the COL applicant if it proposes an onsite low-level radioactive waste storage facility to the NRC. The NRC would then evaluate the proposed facility against the criteria in GL-81-38.

Generic Letter No. and Title	Staff Disposition
GL-82-17, Inconsistency of requirements between 10 CFR 50.54(t) and 50.15	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-82-23, Inconsistency between requirements of 10 CFR 73.40(d) and Standard Technical Specifications (STS) for performing audits of Safeguards Contingency Plans	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-82-39, Problems with submittals of 10 CFR 73.21 safeguards information for licensing reviews	This issue is resolved for the System 80+ design. It is not a design issue. As stated in Section 13.6 of this report, site security is within the scope of the COL applicant. This will include the reporting of safeguards information for licensing reviews.
GL-83-07, The Nuclear Waste Policy Act of 1982	This issue is resolved for the System 80+ design. It is not a design issue. The Nuclear Waste Policy Act of 1982 requires licensees to have a contract with the Department of Energy (DOE) before receiving a license and is within the scope of the COL applicant.
GL-83-09, Review of Combustion Engineering Owners Group Emergency Procedures Guideline Program	This issue is resolved for the System 80+ design. ABB-CE submitted emergency operating procedures guidelines that were reviewed and approved by the staff. They are discussed in Section 18.7 of this report. The COL applicant will use these guidelines to prepare plant-specific emergency operating procedures. See the resolution of Issue I.C.1 in Section 20.3 of this chapter.
GL-83-13, Clarification of surveillance requirements for and charcoal adsorber units in STS on ESF cleanup systems	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-83-26, Clarification of surveillance requirements for diesel fuel impurity level tests	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-83-27, Surveillance intervals in STS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.

Generic Letter No. and Title	Staff Disposition
GL-83-30, Deletion of STS Surveillance Requirement 4.8.1.1.2.D.6 for diesel generator testing	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-84-12, Compliance with 10 CFR Part 61 and implementation of radiological effluent TS, attendant process control program	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-84-13, Technical specifications for snubbers	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-84-21, Long-term, low-power operation in PWRs	This issue is resolved for the System 80+ design. This issue is an operational issue and within the scope of the COL applicant. It is not a design issue.
GL-85-16, High boron concentrations	This issue is resolved for the System 80+ design. It is applicable to Westinghouse-designed plants that use a boron injection tank (BIT); however, it is not applicable to the System 80+ design.
GL-85-19, Reporting requirements on primary coolant iodine spikes	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-86-07, Transmittal of NUREG-1190 regarding the San Onofre Unit 1 loss-of-power and water-hammer event	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-86-13, Potential inconsistency between plant safety analyses and TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-87-09, Sections 3.0 and 4.0 of STS on limiting conditions for operation and surveillance requirements	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-88-02, Integrated Safety Assessment Program II (ISAP II)	This issue is resolved for the System 80+ design. Risk insights are already an integral part of the staff's ongoing System 80+ design review process. See the discussion on severe accidents and PRA for the design in Chapter 19 of this report.

Generic Letter No. and Title	Staff Disposition
GL-88-05, Boric acid corrosion of carbon steel reactor pressure boundary in PWR plant components	This issue is resolved for the System 80+ design. This is a maintenance issue and within the scope of the COL applicant. It is not a design issue.
GL-88-12, Removal of fire-protection requirements from TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-88-15, Electric power systems — inadequate control over design process	This issue is resolved for the System 80+ design. ABB-CE has committed to a number of studies and analyses of the electric power systems. The staff indicated that it would pursue confirmation of the adequacy of these actions as part of ITAACs (DSER Open Item 20.4-2). The adequacy and acceptability of the ABB-CE System 80+ design descriptions in ITAACs are evaluated in Chapter 14 of this report. On the basis of this evaluation, DSER Open Item 20.4-2 is resolved.
GL-88-16, Removal of cycle-specific parameter limits from plant TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-88-18, Plant record storage on optical disks	This issue is resolved for the System 80+ design. This issue is not a design issue. The use of optical disks for storage of records is optional and within the scope of the COL applicant. If this method is selected, it should be addressed in the applicable quality assurance program.
GL-88-20, Individual plant examination for severe-accident vulnerabilities	This issue is resolved for the System 80+ design. Risk insights are already an integral part of the staff's ongoing System 80+ design review process as discussed in Chapter 19 of this report on severe accidents and PRA for the System 80+ design.
GL-89-01, Implementation of programmatic and procedural controls for radiological effluent TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-89-02, Actions to improve the detection of counterfeit and fraudulently marketed products	This issue is resolved for the System 80+ design. This GL is not applicable to the System 80+ design because it involves the procurement of vendor products which is within the scope of the COL applicant. It is not a design issue.
GL-89-07, Power reactor safeguards contingency planning for surface vehicle bombs	This issue is resolved for the System 80+ design. This issue is addressed in Appendix 13A of CESSAR-DC. Industrial security and sabotage protection for the System 80+ design is discussed in Section 13.6 of this report and in the resolution of Issue A-29 in Section 20.1 of this chapter.

Generic Letter No. and Title	Staff Disposition
GL-89-14, Line-item improvements in technical specifications — removal of 3.25 limit on extending surveillance intervals	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-90-02, Alternative requirements for fuel assemblies in the design features section of TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-90-09, Alternative requirements for snubber visual inspection intervals and corrective actions	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-91-01, Removal of the schedule for the withdrawal of reactor vessel material specimens from TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-91-04, Changes in TS surveillance intervals to accommodate a 24-month fuel cycle	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-91-08, Removal of component lists from TS	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-91-09, Modification of surveillance interval, electrical protection assemblies in power supplies, reactor protection system	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-91-15, Operating experience feedback report, solenoid-operated valve problems at U.S. reactors	This issue is resolved for the System 80+ design. See the resolution of Issue I.C.5 in Section 20.3 of this chapter. This issue is also discussed in Section 3.9.3.2 of this report.

Generic Letter No. and Title	Staff Disposition
GL-92-01, Revision 1, Reactor vessel structural integrity	This issue is resolved for the System 80+ design. ABB-CE has met 10 CFR Part 50, Appendices G and H, with substantial margins which ensures reactor vessel structural integrity. See Section 5.3.3 of this report.
GL-92-08, Thermo-Lag 330-1 fire barriers	This is discussed for BL-92-01 in Table 20.3 of this report.
GL-93-05, Line-item TS improvements to reduce surveillance requirements for testing during power operation	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.
GL-93-07, Modification of the TS administrative control requirements for emergency and security plans	This issue is resolved for the System 80+ design. This is an operational issue and not a design issue. These plans are discussed in Sections 13.3 and 13.6, respectively, of this report and are the responsibility of the COL applicant.
GL-93-08, Relocation of TS tables of instrument response time limits	By adoption of the improved ABB-CE STS, ABB-CE has adequately addressed the TS issues in this GL. The ABB-CE TS for the System 80+ design are in CESSAR-DC Chapter 16 and discussed in Chapter 16 of this report.

21 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) for ABB-CE standard designs conducted reviews of issues related to the ABB-CE System 80+ design from April 3, 1990, to May 7, 1994 (see attachment to Appendix E). At the 409th meeting of the full ACRS on May 5, 1994, the Committee considered the ABB-CE application for design certification and issued a letter dated May 11, 1994, to the Chairman of the NRC. This letter is included as Appendix E to this report. No response is required from the applicant or the staff.

22 CONCLUSIONS

The staff performed its review of the CESSAR-DC, certified design material, and technical specifications in accordance with the standards for review of design certification applications set forth in 10 CFR 52.48 that are applicable and technically relevant to the System 80+ standard design, including the exemptions and staff-proposed applicable regulations identified in Section 1.6 of this report. On the basis of its evaluation and independent analyses as discussed in this report, the staff concludes that ABB-CE's application for design certification meets those portions of 10 CFR 52.47 that are applicable and technically relevant to the System 80+ standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR 52.53 is provided in Appendix E of this report.

The staff also concludes that issuance of a final design approval, in accordance with Appendix O to 10 CFR Part 52, will not be inimical to the common defense and security or to the health and safety of the public. The financial qualifications of the applicable utility and the indemnity requirements of 10 CFR Part 140 will be addressed during the plant-specific licensing process for an application that references the System 80+ standard design.

APPENDIX A

CHRONOLOGY OF CORRESPONDENCE

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and ABB-CE regarding the review of the System 80+ design under Project 675 and Docket Number 52-002.

April 2, 1987	J.J. Raleigh, NRC, meeting summary of March 24, 1987, meeting with CE, DOE and EPRI in Bethesda, Maryland regarding certification aspects of CESSAR-DC modified to represent next generation design for future referenceability. FICHE: 40466 025 acn: 8704090118
April 23, 1987	A.E. Scherer, CE, letter providing formal description of CE efforts to advance System 80R PWR design and advises that System 80+ design including consideration of EPRI advanced LWR design requirements document revised. FICHE: 40699 177 acn: 8704280387
June 16, 1987	G.S. Vissing, NRC, meeting summary of June 1, 1987, meeting with CE in Bethesda, Maryland regarding plans for developing PRA for advanced CESSAR-DC design. FICHE: 41416 325 acn: 8706220437
July 1, 1987	Text-safety report - draft "System 80R Design Certification Licensing Basis Agreement." FICHE: 41599 031 acn: 8707070319
July 2, 1987	A.E. Scherer, CE, letter forwarding draft "...Licensing Basis Agreement," for review and comment. FICHE: 41599 029 acn: 8707070191
August 11, 1987	G.S. Vissing, NRC, meeting summary of August 3, 1987, meeting with DOE, IT Corp and CE regarding approach to resolve severe accident issues for advanced LWR program. FICHE: 42169 060 acn: 8708140224
August 19, 1987	G.S. Vissing, NRC, meeting summary of August 12, 1987, meeting with CE in Bethesda, Maryland regarding severe accident issues for System 80+ design. FICHE: 42397 180 acn: 8708280311
September 9, 1987	Text-safety report - "Amendment 12 to CESSAR-F." FICHE: 43570 117 acn: 8712070248
September 11, 1987	Text-safety report - "Amendment 12 to CESSAR FSAR." FICHE: 42817 057 acn: 8709290410

September 18, 1987	A.E. Scherer, CE, letter forwarding Amendment 12 to CESSAR FSAR modifying System 80R design. FICHE: 42817 055 acn: 8709290409
September 18, 1987	A.E. Scherer, CE, letter requesting NRC adoption of proposed docketing process for forthcoming revisions to CE std SAR. FICHE: 42808 143 acn: 8709290234
September 25, 1987	G.S. Vissing, NRC, meeting summary with CE in Bethesda, Maryland regarding USIs & generic issues for System 80+ design. FICHE: 42877 319 acn: 8710010418
October 13, 1987	L.S. Rubenstein, NRC, letter responding to September 18, 1987, letter regarding creation of new docket for CESSAR-DC. FICHE: 43155 241 acn: 8710230061
October 29, 1987	CE, draft "System 80+ TM Design Certification Licensing Review Bases. FICHE: 43342 029 acn: 8711100213
November 10, 1987	G.S. Vissing, NRC, meeting summary of October 29, 1987, meeting with CE in Bethesda, Maryland regarding executive overview of CESSAR-DC System 80+ design certification program. FICHE: 43416 129 acn: 8711230034
November 24, 1987	A.E. Scherer, CE, letter forwarding proposed advanced reactor severe accident program resolutions for four remaining NRC-IDCOR severe accident issues. FICHE: 43476 058 acn: 8712010165
November 24, 1987	G.S. Vissing, NRC, meeting summary of November 19, 1987, meeting with CE in Bethesda, Maryland regarding special features of System 80+ design for design certification. FICHE: 43636 130 acn: 8712090049
November 30, 1987	A.E. Scherer, CE, letter forwarding proposed amendment to CESSAR FSAR describing System 80+ design. FICHE: 43570 115 acn: 8712070222
December 3, 1987	G.S. Vissing, NRC, letter requesting that enclosure material regarding System 80+ should be added to file for docket. FICHE: 43589 298 acn: 8712080316
December 7, 1987	G.S. Vissing, NRC, forwarding comments to aid in redrafting licensing review bases for System 80+ design for design certification, provided during October 29, 1987, meeting. FICHE: 43661 130 acn: 8712100387

December 8, 1987	G.S. Vissing, NRC, letter forwarding request for additional information regarding Chapter 1, Amendment 12 of System 80+ CESSAR-DC FICHE: 43703 310 acn: 8712140235
December 8, 1987	G.S. Vissing, NRC, letter discusses November 19, 1987, CESSAR-DC review kickoff meeting regarding safeguards considerations for System 80+ design. FICHE: 43690 351 acn: 8712140305
December 17, 1987	G.S. Vissing, NRC, letter forwarding Reactor Safeguards Branch comments and request for additional information regarding Chapter 1, "Safeguards," Amendment 12 to System 80+, CESSAR-DC. FICHE: 43786 058 acn: 8712230071
December 17, 1987	G.S. Vissing, NRC, letter forwarding request for additional information regarding Chapter 1, Amendment 12 to System 80+ CESSAR-DC. FICHE: 43785 224 acn: 8712230136
January 11, 1988	G.S. Vissing, NRC, letter forwarding NUREG-0852, Supplement 3, SER regarding final design of NSSS for CESSAR System 80+. FICHE: 44066 253 acn: 8801190214
January 13, 1988	G.S. Vissing, NRC, notification of January 26, 1988, meeting with CE in Bethesda, Maryland to discuss base-line PRA for System 80+ design CESSAR-DC. FICHE: 44119 172 acn: 880125072
January 13, 1988	C.J. Holloway, NRC, letter advises that whether System 80+ considered extension to FDA-2 or approved s separate FDA for design certification, system subject to full cost recovery payable in 20 percent increments no later than 10 years from certification issuance date. FICHE: 44055 171 acn: 8801190233
January 15, 1988	G.S. Vissing, NRC, meeting summary of December 16, 1987, meeting with Intl. Technology Corp. in Bethesda, Maryland regarding DOE advanced reactor severe accident program. FICHE: 44180 129 acn: 8801290161
January 18, 1988	Text-safety report - draft "CE System 80+ Tm std Design." FICHE: 44110 003 acn: 8801250377
January 19, 1988	A.E. Scherer, CE, letter forwarding input to licensing review bases document being developed for CESSAR-DC originally transmitted on July 2, 1987. FICHE: 44110 001 acn: 8801250367

January 22, 1988	A.E. Scherer, CE, letter forwarding proprietary "Base Line Level 1," for System 80R NSSS Design. FICHE: 44152 319 acn: 8801280263
January 25, 1988	L.S. Rubenstein, NRC, letter forwarding request for additional information regarding design goals addressing severe accidents, based on review of advanced LWR application. FICHE: 44296 030 acn: 8802090426
January 31, 1988	Text-safety report - "Nonproprietary 'Base Line Level 1' PRA for System 80R NSSS Design." FICHE: 44508 314 acn: 8802260246
February 16, 1988	A.E. Scherer, CE, letter forwarding proprietary and nonproprietary 'Base Line Level 1' PRA for System 80R NSSS Design,' per NRC request. FICHE: 8802260187 acn: 44508 312
February 18, 1988	G.S. Vissing, NRC, meeting summary of January 26, 1988, meeting with CE in Bethesda, Maryland regarding base line PRA for System 80+ design. FICHE: 44499 210 acn: 8802250150
February 25, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding vendor November 30, 1987, submittal of Amendment 12, Chapter 10 of System 80+ CESSAR-DC. FICHE: 44540 356 acn: 8803010289
February 26, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Topical Report CENPD-210A, Revision 4, "QA Program" Chapter 17 of CESSAR-DC. FICHE: 44553 356 acn: 8803030095
March 11, 1988	G.S. Vissing, NRC, letter approves January 22, 1988, request to withhold base line level 1 PRA for System 80+ NSSS design from public disclosure. FICHE: 44689 295 acn: 8803150040
March 11, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment 12, Chapters 1 and 10 of System 80+ CESSAR-DC, transmitted by November 30, 1987, letter. FICHE: 44725 320 acn: 8803170224
March 15, 1988	G.S. Vissing, NRC, letter requesting for additional information regarding Chapter 10 of Plant System Branch CESSAR-DC. FICHE: 44746 357 acn: 8803210450
March 15, 1988	L.S. Rubenstein, NRC, letter informs of relocation of NRR to stated address in Rockville, Maryland. FICHE: 44760 138 acn: 8803210550

March 18, 1988	A.E. Scherer, CE, letter forwarding additional information regarding chemical and volume control system for System 80+ TM std design per December 17, 1987, request. FICHE: 44912 152 acn: 8803280316
March 22, 1988	A.E. Scherer, CE, letter forwarding response to NRC December 8, 1987, request for additional information regarding regulatory guides that appear in CESSAR-DC QA. FICHE: 44912 029 acn: 8803280228
April 11, 1988	Text-safety report - " CESSAR-DC Submittal Group B - Revisions to Chapters 1, 4, 5 and 9." FICHE: 45180 149 acn: 8804200542
April 11, 1988	A.E. Scherer, CE, letter forwarding draft revisions to CE Chapters 1, 4, 5 and 9 to CE standard SAR. FICHE: 45180 131 acn: 8804200529
April 13, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment 12, Chapter 1 of System 80+ CESSAR-DC transmitted by September 11, 1987, letter. FICHE: 45200 235 acn: 8804210448
April 14, 1988	L.S. Rubenstein, NRC, letter forwarding comments on vendor January 19 and March 2, 1988, submittals regarding CESSAR-DC System 80+ design. FICHE: 45264 049 acn: 8804280275
May 12, 1988	G.S. Vissing, NRC, notification of June 1, 1988, meeting with CE in Rockville, Maryland to discuss proposed changes to CESSAR-DC System 80+ QA Program. FICHE: 45585 176 acn: 8805200336
May 25, 1988	A.E. Scherer, CE, letter providing additional information on CESSAR-DC Chapter 10 regarding secondary water chemistry per NRC February 25, 1988, request. FICHE: 45733 075 acn: 8806010034
May 25, 1988	A.E. Scherer, CE, letter forwarding response to NRC February 26, 1988, request for additional information on Chapter 17 to Revision 4 to topical report CENPD-210A, "QA Program," including revisions to CE November 30, 1987 and April 11, 1988 responses. FICHE: 45718 340 acn: 8806010249
May 31, 1988	Text-Safety report - Nonproprietary "Functional Design Requirements for Control Element Assembly Calculator." FICHE: 60058 227 acn: 9112190178
May 31, 1988	Text-Safety report - "Functional Design Requirements for Core Protection Calculator." FICHE: 60058 307 acn: 9112190179

June 1, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Chapters 1 an 5 of Amendment B of System 80+ CESSAR-DC. FICHE: 45796 186 acn: 8806130263
June 2, 1988	G.S. Vissing, NRC, notification of June 28, 1988, meeting with CE in Rockville, Maryland to discuss proposed changes to CESSAR-DC System 80+ QA program. FICHE: 45744 211 acn: 8806070400
June 6, 1988	A.E. Scherer, CE, letter providing proposed resolutions for four of six issues which make up topic paper set 2 including in-vessel hydrogen generation core melt progression and vessel failure and hydrogen ignition and burning. FICHE: 45840 237 acn: 8806160126
June 6, 1988	A.E. Scherer, CE, letter forwarding additional information regarding Chapters 1 and 10 to CESSAR-DC per NRC March 11, 1988, request. FICHE: 45835 334 acn: 8806150430
June 17, 1988	A.E. Scherer, CE, letter providing proposed resolutions to remaining two issues committed in June 6, 1988, letter regarding direct containment heating and debris coolability. FICHE: 45962 142 acn: 8806290214
June 20, 1988	G.S. Vissing, NRC, meeting summary of May 19, 1988, meeting with CE in Rockville, Maryland regarding human factors program in control room design for CESSAR-DC System 80+. FICHE: 46113 179 acn: 8807120374
June 28, 1988	G.S. Vissing, NRC, letter requesting additional information regarding Amendment B of Chapters 5 and 9 of CESSAR-DC, System 80+. FICHE: 46113 356 acn: 8807120720
June 28, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment B of Chapters 4 and 5 of CESSAR-DC System 80+. FICHE: 46120 200 acn: 8807120716
June 30, 1988	A.E. Scherer, CE, letter forwarding response to NRC March 15, 1988 request for additional information regarding CESSAR-DC Chapter 10, "Plant System Branch." FICHE: 46185 043 acn: 8807140271
June 30, 1988	Text-safety report - "Amendment C to CESSAR-DC." FICHE: 46157 207 acn: 8807120199
June 30, 1988	A.E. Scherer, CE, letter forwarding Amendment C to CESSAR-DC Chapters 5, 6 and 10. FICHE: 46157 196 acn: 8807120188

July 1, 1988	Text-safety report - "Flow Distribution and Tube Vibration: Evaluation of System 80+ Steam Generator Tube Lane-Economizer Corner Region." FICHE: 46182 094 acn: 8807180001
July 1, 1988	G.S. Vissing, NRC, meeting summary of June 28, 1988, meeting with CE in Rockville, Maryland regarding QA program plan for CESSAR-DC System 80+, request for additional information regarding review of Chapter 17 of Amendment 12 and engineering plans for revising QA program plan. FICHE: 46184 063 acn: 8807140097
July 1, 1988	A.E. Scherer, CE, letter forwarding "Flow Distribution and Tube Vibration: Evaluation of System 80 Steam Generator Tube Lane-Economizer Corner Region," in support of CE September 18 1987 request that steam generator tube vibration issue be closed. FICHE: 46182 092 acn: 8807150268
July 15, 1988	A.E. Scherer, CE, letter forwarding revised "Design Certification Licensing Review Basis." FICHE: 46314 249 acn: 8807280126
July 15, 1988	Text-safety report - "Design Certification Licensing Review Bases." FICHE: 46314 250 acn: 8807280140
July 28, 1988	G.S. Vissing, NRC, meeting summary of June 21, 1988, meeting with CE and Intl. Technology Corp. in Rockville, Maryland regarding ARSAP topic papers sets 1 and 2. FICHE: 46452 201 acn: 8808080208
July 29, 1988	A.E. Scherer, CE, letter forwarding DOE advanced reactor severe accident program proposed resolutions for severe accident issues topic set 3. FICHE: 46560 226 acn: 8808110042
August 1, 1988	A.E. Scherer, CE, letter forwarding response to request for additional information regarding CESSAR-DC Chapters 1 and 5 regarding sabotage protection per June 1, 1988, letter. FICHE: 46485 246 acn: 8808100308
August 2, 1988	A.E. Scherer, CE, letter informing of finalizing review of QA program description of Revision 5 to CESSAR topical report CENPD-210 and anticipates transmitting review to NRC by third quarter 1988. FICHE: 46792 183 acn: 8809080061
August 2, 1988	G.S. Vissing, NRC, letter discussing QA for CESSAR-DC System 80+ and significant points made listed. FICHE: 46525 333 acn: 8808120030

August 2, 1988	A.E. Scherer, CE, letter informing that vendor finalizing review and anticipates that Revision 5 to CENPD-210 will be transmitted by September 1988. FICHE: 46555 317 acn: 8808110311
August 3, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Chapter 9, "Auxiliary Systems" and Chapter 5, "RCS" CESSAR-DC System 80+ in order to continue review of Amendment B. FICHE: 46469 018 acn: 8808080184
September 9, 1988	A.E. Scherer, CE, letter forwarding additional information regarding CESSAR-DC Chapter 4 in response to NRC June 28, 1988 request. FICHE: 46908 355 acn: 8809190298
September 9, 1988	A.E. Scherer, CE, letter providing proposed resolution for single issue of advanced reactor severe accident program topic paper set 4, "Essential Equipment Performance. FICHE: 46982 130 acn: 8809290307
September 12, 1988	L.W. Zech, NRC, letter responding to comments and questions regarding regulatory trends in U.S. an Republic of Korea. FICHE: 46943 008 acn: 8809230196
September 12, 1988	K.J. Shik, Korea, letter submitting questions from South Korean engineers regarding value of design certification program for CE System 80 and System 80+ designs. FICHE: 46943 010 acn: 8809230199
September 14, 1988	A.E. Scherer, CE, letter forwarding summary of sabotage protection considerations and draft requirements for sabotage design from EPRI advanced LWR requirements document per NRC December 8, 1987, letter. FICHE: 46920 112 acn: 8809210116
September 20, 1988	A.E. Scherer, CE, letter responding to NRC request for additional information regarding Chapters 5 and 9 of CESSAR-DC regarding steam generator secondary water chemistry, reactor coolant water chemistry, fire protection system, letdown purification line, and hydrogen ignition. FICHE: 46983 249 acn: 8809290014
September 30, 1988	A.E. Scherer, CE, letter forwarding Amendment D to CESSAR FSAR including revisions to Chapters 2 through 7 and 18. FICHE: 47090 001 acn: 8810110320
September 30, 1988	Text-Safety report - "Amendment D to FSAR for CESSAR." FICHE: 47090 006 acn: 8810110356

October 11, 1988	G.S. Vissing, NRC, letter requesting additional information consisting of "Nuclear Fission Product Aerosol Transport and Deposition," to review ARSAP topic paper Set 1. FICHE: 47146 270 acn: 8810140083
October 20, 1988	G.S. Vissing, NRC, letter requesting additional information regarding Chapter 10, "Auxiliary Systems," of CESSAR-DC System 80+. FICHE: 47342 292 acn: 8810280075
October 21, 1988	A.E. Scherer, CE, letter responding to NRC request for additional information on Chapter 17 of CESSAR-DC. FICHE: 47411 289 acn: 8811020103
October 26, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding September 30, 1988, submittal of Amendment D to Chapters 4, 5, 6 and 10 of CESSAR-DC System 80+. FICHE: 47412 042 acn: 8811020247
October 28, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Chapter 5 Amendment C to CESSAR-DC System 80+ on steam generators. FICHE: 47439 355 acn: 8811040166
November 1, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment C of Chapters 5, 6 and 10 of CESSAR-DC System 80+. FICHE: 47440 108 acn: 8811040272
November 2, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment D of Chapter 7 of CESSAR-DC, System 80+ transmitted with CE September 30, 1988, letter, including physical separation, trip function calculation and remote shutdown panel controls. FICHE: 47494 277 acn: 8811080163
November 3, 1988	G.S. Vissing, NRC, meeting summary of September 28, 1988, meeting with CE and IT Corp in Rockville, Maryland regarding ARSAP topic papers set 3. FICHE: 47513 145 acn: 8811090364
November 4, 1988	A.E. Scherer, CE, letter forwarding QA program in response to NRC request for additional information regarding Chapter 17 CESSAR-DC QA. FICHE: 47569 129 acn: 8811160402
November 11, 1988	A.E. Scherer, CE, letter forwarding advanced reactor severe accident program topic paper set 6, "Development of Severe Accident Management Program," for review. FICHE: 47680 222 acn: 8811300491

November 21, 1988	A.E. Scherer, CE, letter submitting guidance regarding scope of design and scope of staff review of CESSAR-DC, System 80++. FICHE: 47649 265 acn: 8811290064
November 28, 1988	G.S. Vissing, NRC, notification of December 20, 1988, meeting with CE in Rockville, Maryland to discuss seismic issues involved in containment design for CESSAR-DC System 80E regarding soil-structure interaction phase. FICHE: 47732 300 acn: 8812050189
December 7, 1988	A.E. Scherer, CE, letter forwarding additional information regarding CESSAR-DC Chapters 5 and 9 per G.S. Vissing August 3, 1988 request. FICHE: 47802 329 acn: 8812150015
December 15, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding ARSAP topic paper 3 including Item 3.1. "External Events," Item 3.2, "Success Criteria and Mission Time" and Item 3.3, "Accident Sequence Selection." FICHE: 47900 220 acn: 8812220242
December 16, 1988	G.S. Vissing, NRC, letter forwarding requests for additional information regarding Amendment D to Chapters 3, 4, 5 and 6 of CESSAR-DC System 80+. FICHE: 47900 209 acn: 8812220286
December 23, 1988	A.E. Scherer, CE, letter forwarding Amendment E to CESSAR FSAR including revisions to Chapters 1, 3, 6, 7, 9, 12, 13, 14, 17 and 18, regarding design certification summary of revisions. FICHE: 47939 132 acn: 8812280055
December 23, 1988	A.E. Scherer, CE, letter forwarding resolutions for items in topic paper set 5 'Advanced Reactor Severe Accident Program,' consisting of implementation of NRC safety goal policy uncertainties in plant risk and acceptability of MAAP-DOE code. FICHE: 48027 323 acn: 8901040017
December 23, 1988	G.S. Vissing, NRC, letter requesting additional information on September 30, 1988, Chapters 4 and 6 of Amendment D to CESSAR-DC regarding functional design of reactivity control system and ECCS design, respectively. FICHE: 48106 318 acn: 8901090336
December 23, 1988	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment D, Chapters 2 through 7 and 18 of CESSAR-DC, System 80+. FICHE: 48137 140 acn: 8901110218
December 30, 1988	Text-safety report - "Amendment E to CESSAR-F standard SAR Design Certification." FICHE: 47939 136 acn: 8812280061

December 31, 1988	Text-safety report - "Volume VIII to 'Resolution of Outstanding Nuclear Fission Product Aerosol Transport and Deposition Issues WBS 3.4.2.'" FICHE: 49058 099 acn: 8903270045
January 1, 1989	Text-safety report - "System 80+ Function and Task Analysis Final Report." FICHE: 61781 175 acn: 9205210192
January 19, 1989	G.S. Vissing, NRC, letter forwarding request for additional information regarding Amendment D of Chapters 2, 3, 5 and 6 of CESSAR-DC FSAR in respond to September 30, 1989, application. FICHE: 48313 060 acn: 8901310387
February 28, 1989	D.G. Harrison, TENERA, "Advanced Reactor Severe Accident Program, CE interim external events Integration WBS 10.4.4." FICHE: 50547 183 acn: 8907170102
March 15, 1989	A.E. Scherer, CE, letter forwarding Amendment E to CE standard SAR Group E2 regarding design certification. FICHE: 49040 238 acn: 8903240108
March 19, 1989	T.J. Kenyon, NRC, letter forwarding comments regarding CESSAR-DC baseline PRA submitted by January 22, 1988, letter. FICHE: 48983 188 acn: 8903210414
March 27, 1989	T.J. Kenyon, NRC, meeting summary of February 7, 1989, meeting in Rockville, Maryland regarding licensing issues for future advanced LWRs. FICHE: 69679 067 acn: 8903100342
March 30, 1989	Text-safety report - Appendix 3.11B, "Identification and Location of Mechanical and Electrical Safety-Related System Components," to CESSAR System 80+ standard design. FICHE: 49252 150 acn: 8904110188
March 30, 1989	Text-safety report - Appendix 3.11A, "Environmental Qualification for Structures and Components," to CESSAR System 80+ standard design. FICHE: 49252 128 acn: 8904110181
March 30, 1989	Text-safety report - Chapter 18, "Human Factors Engineering," to CESSAR System 80+ standard design. FICHE: 49263 041 acn: 8904050008
March 30, 1989	Text-safety report - Chapter 17, "QA Program," to CESSAR System 80+ standard design. FICHE: 49263 037 acn: 8904040465

March 30, 1989	Text-safety report - Chapter 16, "Tech Specs," to CESSAR System 80+ standard design. FICHE: 49262 076 acn: 8904040464
March 30, 1989	Text-safety report - Appendix 15D, "Steam Generator Tube Rupture with Loss of Offsite Power and Single Failure," to CESSAR System 80+ standard design. FICHE: 49262 022 acn: 8904070072
March 30, 1989	Text-safety report - Appendix 15C,"Analysis Methods for Steamline Breaks, to CESSAR System 80+ standard design. FICHE: 49262 002 acn: 8904040462
March 30, 1989	Text-safety report - Appendix 15B, "Methods for Analysis of Loss of Feedwater Inventory Events," to CESSAR System 80+ standard design. FICHE: 49261 291 acn: 8904040460
March 30, 1989	Text-safety report - Appendix 15A, "Loss of Primary Coolant Flow Methodology Description," to CESSAR System 80+ standard design. FICHE: 49261 271 acn: 8904040457
March 30, 1989	Text-safety report - Chapter 8, "Electrical Power," to CESSAR System 80+ standard design. FICHE: 49257 208 acn: 8904040437
March 30, 1989	Text-safety report - Chapter 15, "Accident Analyses," to CESSAR System 80+ standard design. FICHE: 49260 034 acn: 8904040456
March 30, 1989	Text-safety report - Chapter 7, "Instrumentation and Controls," to CESSAR System 80+ standard design. FICHE: 49256 107 acn: 8904040434
March 30, 1989	Text-safety report - Chapter 14, "Initial Test Program," to CESSAR System 80+ standard design. FICHE: 49259 238 acn: 8904040452
March 30, 1989	Text-safety report - Appendix 13A, "Sabotage Protection," to CESSAR System 80+ standard design. FICHE: 49259 215 date: 890330
March 30, 1989	Text-safety report - Chapter 6, "ESFS," to CESSAR System 80+ standard design. With one oversize enclosure. FICHE: 49254 089 acn: 8904040375

March 30, 1989	Text-safety report - Chapter 13, "Conduct of Operators," to CESSAR System 80+ standard design. FICHE: 49259 191 acn: 8904040450
March 30, 1989	Text-safety report - Chapter 12, "Radiation Protection," to CESSAR System 80+ standard design. FICHE: 49259 163 acn: 8904040449
March 30, 1989	Text-safety report - Appendix 11A, "Core Residence Times," to CESSAR System 80+ standard design. FICHE: 49259 157 acn: 8904040445
March 30, 1989	Text-safety report - Appendix 5C, "Structural Evaluation of Feedwater Line Break for Steam Generator Internals," to CESSAR System 80+ standard design. FICHE: 49254 079 acn: 8904040360
March 30, 1989	Text-safety report - Chapter 11, "Radwaste Management," to CESSAR System 80+ standard design. Fiche: 49259 088 acn: 8904040443
March 30, 1989	Text-safety report - Appendix 5B, "Structural Evaluation of Steamline Break for Steam Generator Internals," to CESSAR System 80+ standard design. FICHE: 49254 070 acn: 8904040358
March 30, 1989	Text-safety report - Appendix 5A, "Overpressure Protection for CE System 80 PWRS," to CESSAR System 80+ standard design. FICHE: 49254 054 acn: 8904040347
March 30, 1989	Text-safety report - Appendix 10A, "Emergency Feedwater System Reliability Analysis," to CESSAR System 80+ standard design. FICHE: 49259 033 acn: 8904070042
March 30, 1989	Text-safety report - Chapter 5, "RCS and Connected System," to CESSAR System 80+ standard design. With 2 oversize enclosures. FICHE: 49253 198 acn: 8904040345
March 30, 1989	Text-safety report - Appendix 4B, "Hot LOOP Flow Testing of System 80+ Fuel and Control Element Assembly Components," to CESSAR 80 standard design. FICHE: 49253 186 acn: 8904040335
March 30, 1989	Text-safety report - Appendix 4A, "System 80+ Reactor Flow Model Test Program," to CESSAR System 80+ standard design. FICHE: 49253 168 acn: 8904040334

March 30, 1989

Text-safety report - Chapter 10, "Steam and Power Conversion System," to CESSAR System 80+ standard design. With 1 oversize enclosure.
FICHE: 49258 251
acn: 8904040441

March 30, 1989

Text-safety report - Chapter 4, "Reactor," to CESSAR System 80+ standard design.
FICHE: 49252 175
acn: 8904040332

March 30, 1989

Text-safety report - Chapter 9, "Auxiliary System," to CESSAR System 80+ standard design.
FICHE: 49257 302
date: 890330

March 30, 1989

Text-safety report - Chapter 3, "Design of Structures Components Equipment and System," to CESSAR System 80+ standard design.
FICHE: 49251 148
acn: 8904040304

March 30, 1989

Text-safety report - Chapter 2, "Site Envelope Characteristics," to CESSAR System 80+ standard design.
FICHE: 49251 134
acn: 8904040291

March 30, 1989

Text-safety report - Chapter 1, "Introduction and General Plant Description," to CESSAR System 80+ standard design. With 1 oversize enclosure.
FICHE: 49251 009
acn: 8904040288

March 30, 1989

Text-safety report - Volumes 1 through 17 consisting of Chapters 1 through 18 to "CESSAR Design Certification" through Amendment E.
FICHE: 49251 004
acn: 8904040284

March 30, 1989

A.E. Scherer, forwarding Volumes 1 through 17 consisting of Chapters 1 through 18 to "CESSAR Design Certification" for approval per 10 CFR Part 52.
FICHE: 49251 001
acn: 8904040261

March 30, 1989

A.E. Scherer, CE, letter forwarding "Design Certification Licensing Review Basis," for review and concurrence.
FICHE: 49294 278
acn: 8904130024

March 31, 1989

CE, "Design Certification Licensing Review Basis."
FICHE: 49294 280
acn: 8904130025

May 1, 1989

A.E. Scherer, CE, letter forwarding response to request for additional information regarding CESSAR-DC, Chapter 5.
FICHE: 49899 044
acn: 8905230116

June 26, 1989	T.J. Kenyon, NRC, letter forwarding request for additional information regarding CESSAR-DC, System 80+ including emergency preparedness, plant system, reactor system, chemistry, radiation protection and reactor safeguards. FICHE: 50530 231 acn: 8907130132
June 30, 1989	C.D. Gentillon, forwards draft "Component Failure Data Handbook" technical report. FICHE: 70031 002 acn: 8910250036 acn: 9201290130
July 6, 1989	T.J. Kenyon, NRC, meeting summary of June 6 and 7, 1989, meeting with BNL and CE regarding concerns resulting from NRC and BNL review of System 80+ baseline PRA. FICHE: 50541 343 acn: 89707140157
July 7, 1989	A.E. Scherer, CE, letter forwarding response to NRC December 15, 1988, request for additional information regarding advanced reactor severe accident program topic paper set 3 involving PRA methodology. FICHE: 50547 171 acn: 8907170039
July 21, 1989	A.E. Scherer, CE, letter forwarding response to November 1, 1988, request for additional information regarding CESSAR-DC, Chapter 10, "Chemical Engineering." FICHE: 50676 212 acn: 8907260023
August 7, 1989	T.E. Murley, NRC, provides clarification and further guidance regarding containment design to assure that containment conditional failure probability less than 1 in 10 when weighted over credible core damage sequences. Goal of 0.1 possible. FICHE: 50916 133 acn: 8908140099
August 7, 1989	A.E. Scherer, CE, letter forwarding "CE System 80+ standard Design, Design Certification Licensing Review Basis." FICHE: 50927 337 acn: 8908160261
August 21, 1989	A.E. Scherer, CE, letter confirms that CE application for design certification for System 80+ standard design considered submitted per 10 CFR 52.45. FICHE: 51074 232 acn: 8908250282
August 21, 1989	A.E. Scherer, CE, letter confirming that CE application for design certification for System 80+ standard design considered submitted per 10 CFR 52.45. FICHE: 51074 232 acn: 8908250282
August 31, 1989	CE, "CE System 80+ Standard Design, Design Certification Licensing Review Basis." FICHE: 50927 388 acn: 8908160264

September 28, 1989	A.E. Scherer, CE, letter forwarding additional information regarding CESSAR-DC Chapter 7. FICHE: 51372 251 acn: 8910030174
October 4, 1989	A.E. Scherer, CE, letter responding to NRC request for additional information regarding CESSAR-DC certification, Chapters 4 and 5. FICHE: 51520 183 acn: 8910160015
October 30, 1989	A.E. Scherer, CE, letter forwarding response to request for additional information regarding CE QA program and CESSAR-DC, Chapter 17, proposed revision to CESSAR-DC and Revision 5 to CENPD-120. FICHE: 51739 067 acn: 8911080004
November 15, 1989	R.N. Singh, NRC, meeting summary of October 20, 1989, meeting with CE in Windsor, Connecticut regarding System 80+ design certification. FICHE: 51856 266 acn: 8911280103
November 21, 1989	CE, letter requesting that NRC provide consent with respect to ref. licenses on or before December 13, 1989, proposed date for consummation of transaction with CE will become wholly owned subsidiary of ABB-CE. FICHE: 5654 075 acn: 9101300116
November 21, 1989	CE, letter requesting NRC consent regarding Topical Report FDA-2 in connection with proposed transaction which CE will become wholly owned subsidiary of ABB-CE. FICHE: 56554 087 acn: 9101300179
December 22, 1989	A.E. Scherer, CE, letter forwarding Amendment F to CE standard SAR - Design Certification. FICHE: 52112 039 acn: 8912270353
December 31, 1989	Text-safety report - "Nuplex 80+ Revision 2 to Verification Analysis Report." FICHE: 61783 012 acn: 9205210194
January 5, 1990	R.M. Burt, CE, letter requesting that consolidated financial statements submitted to NRC by November 21, 1989, letter regarding indirect transfer of CE licenses be treated as confidential, per 10 CFR 2.790. FICHE: 52490 294 acn: 9001300096
January 22, 1990	A.E. Scherer, CE, letter forwarding suggested revision to "Design Certification Licensing Review Basis Document," per January 4, 1990, meeting. FICHE: 70612 005 acn: 9012130040

January 24, 1990	R. Singh, NRC, letter forwarding request for additional information to complete review of System 80+ design certification for CESSAR-DC, including fire protection analysis, fuel assembly storage capacity, storage densities for spent fuel pool and spent fuel pool storage racks. FICHE: 52523 055 acn: 9002010306
January 25, 1990	A.E. Scherer, CE, letter forwarding response to December 16, 1988, request for additional information regarding CESSAR-DC, Chapters 3, 4, 5 and 6 regarding turbine missiles, control element drive structural materials, cleaning and contamination protection procedures and reactor internals materials. FICHE: 52599 194 acn: 9002090070
January 31, 1990	CE, "Design Certification Licensing Review Basis Document.". FICHE: 70612 006 acn: 9012130044
April 26, 1990	E.H. Kennedy, CE, letter forwarding Amendment F to CESSAR-DC and affidavit, per 10 CFR 50.4(b) and 50.30(b). FICHE: 53636 264 acn: 9004300116
April 26, 1990	CE, Amendment F to CESSAR-DC. FICHE: 5636 266 acn: 9004300126
April 26, 1990	Text-safety report - Amendment F to CESSAR-DC. Fiche: 53636 266 acn: 9004300126
April 26, 1990	E.H. Kennedy, letter forwarding Amendment F to CESSAR-DC and affidavit per 10 CFR 50.4(B) and 50.30(B). FICHE: 53636 264 acn: 9004300116
May 18, 1990	E.H. Kennedy, NRC, letter advising that licensee will submit application for final design approval and design certification of process inherent ultimate safety reactor in FY92 and that safe integral reactor anticipated FY93, in addition to System 80+ under review. FICHE: 54372 002 acn: 9006290201
June 29, 1990	R.N. Singh, NRC, meeting summary of May 20 and 21, 1990, meetings with CE in Windsor, Connecticut to discuss instrumentation and control for System 80+. FICHE: 54500 289 acn: 9007120091
July 12, 1990	E.H. Kennedy, CE, letter forwarding additional copies of Amendment G to CESSAR-DC. FICHE: 54714 169 acn: 9007260010
July 12, 1990	E.H. Kennedy, letter forwarding additional copies of Amendment G to CESSAR-DC. FICHE: 54714 169 acn: 9007260010

August 31, 1990	CE, Amendment H to "CESSAR-DC." FICHE: 55432 244 & 55719 131 acn: 9010120206 & 9011070054
August 31, 1990	Text-safety report - Amendment H to "CESSAR - Design Certification." FICHE: 55719 131 acn: 9011070054
August 28, 1990	E.H. Kennedy, CE, letter forwarding proposed changes to System 80+ licensing review basis document, per January 22, 1990, letter. FICHE: 55251 020 acn: 9009190208
October 3, 1990	E.H. Kennedy, CE, letter forwarding Amendment H to "CESSAR-DC." FICHE: 55432 241 acn: 9010120205
October 29, 1990	E.H. Kennedy, CE, letter forwarding Amendment H to "CESSAR-DC." FICHE: 55719 129 acn: 9011070047
October 29, 1990	E.H. Kennedy, letter forwarding Amendment H to "CESSAR - Design Certification." FICHE: 55719 129 acn: 9011070047
November 6, 1990	C.L. Miller, NRC, letter requesting description in specific detail for items listed in Appendix A to licensing review basis document regarding System 80+. FICHE: 55808 229 9011150215
November 13, 1990	T.V. Wambach, NRC, meeting summary of October 3, 1990, meeting with CE regarding CE System 80+ seismic and containment design, structural model development and soil-structure interaction. FICHE: 70603 048 acn: 9012050161
November 23, 1990	T.V. Wambach, NRC, meeting summary of October 15 through 17, 1990, meeting with CE in Windsor, Conn. regarding Chapter 18 of CESSAR-DC System 80+. FICHE: 70588 320 acn: 9011300217
December 3, 1990	C. Michelson, ACRS, meeting summary of ACRS 367th meeting on November 8 through 10, 1990, regarding SECY-90-353, licensing review basis document for CE, System 80+ evolutionary LWR (report dated November 14, 1990) and NUREG-1150, "Severe Accident Rists: Assessment of Five US Nuclear Power Plants." FICHE: 56362 286 acn: 9101080113
December 12, 1990	D.M. Crutchfield, NRC, letter forwarding partial request for additional information regarding CESSAR-DC, System 80+. FICHE: 56244 326 acn: 9101020260

December 20, 1990	D.M. Crutchfield, NRC, letter forwarding partial request for additional information regarding CESSAR-DC, System 80+. FICHE: 56244 326 acn: 9101020260
December 21, 1990	CE, Amendment I to "CESSAR - Design Certification CESSAR-DC)." FICHE: 57013 031 acn: 9103130321
December 24, 1990	T.V. Wambach, NRC, letter forwarding request for additional information on DC application for CE System 80+ design Project 675. FICHE: 56251 049 acn: 9101020300
January 2, 1991	J.C. Hoyle, ACRS, letter forwarding reports and background papers of ACRS for placement in Advisory Committee depository. FICHE: 56362 283 acn: 9101080064
January 30, 1991	E.H. Kennedy, CE, letter forwarding summary of Amendment I to standard for design certification for information and planning purposes. FICHE: 57411 356 acn: 9104160004
January 31, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on CESSAR-DC, System 80+. FICHE: 56767 296 acn: 9102210124
January 31, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on CESSAR-DC, System 80+ based on NRC review of Section 15.4 regarding reactivity accidents. FICHE: 56717 146 acn: 9102150134
February 15, 1991	T.V. Wambach, NRC, letter forwarding request for additional information regarding Chapter 5 and 6 of CESSAR-DC, System 80+ design, based on NRC review. FICHE: 56802 269 acn: 9102250363
February 21, 1991	D.M. Crutchfield, NRC, letter notifies applicants for standard plant design certification of requirement to inform NRC of plans to consider severe accident mitigation design alternatives for proposed designs. FICHE: 56825 269 acn: 9102280139
February 21, 1991	D. Crutchfield, NRC, letter discussing severe accident mitigation design alternatives for certified standard designs. Licensees to inform NRC regarding plans to consider severe accident mitigation design alternatives for proposed designs. FICHE: 56857 178 acn: 9102280120

March 1, 1991	T.V. Wambach, NRC, meeting summary of February 7, 1991, meeting with CE in Rockville, Maryland regarding demonstration of use of proven technology or need for prototype testing of advanced control complex (Nuplex 80+). FICHE: 57036 343 acn: 9103140237
March 4, 1991	E.H. Kennedy, CE, letter responding to NRC December 23, 1988, request for additional information regarding CESSAR-DC. FICHE: 56983 035 acn: 9103120463
March 4, 1991	E.H. Kennedy, CE, letter forwarding Amendment I to "CESSAR - Design Certification (CESSAR-DC)." FICHE: 57013 001 acn: 9103130320
March 15, 1991	E.H. Kennedy, CE, letter forwarding response to NRC January 19, 1989, request for additional information to enable NRC to continue review of CESSAR - design certification. FICHE: 57128 267 acn: 9103220375
March 15, 1991	E.H. Kennedy, CE, letter forwarding response to NRC June 26, 1989, request for additional information regarding CE std SAR - design certification, including revision to CESSAR-DC. FICHE: 57138 118 acn: 9103250292
March 26, 1991	E.H. Kennedy, CE, letter forwarding response to NRC request for additional information regarding design certification, CESSAR-DC, per January 24, 1990, letter. FICHE: 57190 176 acn: 9104010154
March 30, 1991	Text-safety report - Appendix 3A "Discussion of Finite Difference Analysis for Analysis of Pipe Whip" to CESSAR System 80+ standard design. FICHE: 49251 302 acn: 8904100310
April 2, 1991	R.W. Borchardt, NRC, meeting summary of April 2, 1991, meeting with NUMARC regarding implementation of 10 CFR Part 52. FICHE: 57316 057 acn: 9104090251
April 12, 1991	E.H. Kennedy, CE, letter forwarding response to NRC December 21, 1990, request for additional information regarding SSAR for design certification (CESSAR-DC). FICHE: 57437 001 acn: 9104190057
April 26, 1991	E.H. Kennedy, letter forwarding response to NRC request for additional information regarding CESSAR design certification. FICHE: 57645 053 acn: 9105080052

April 30, 1991	C.D. Gentillon, forwards text - procurement and contracts "Component Failure Data Handbook" technical evaluation report. FICHE: 60515 007 acn: 9202110028
May 1, 1991	C.L. Miller, forwarding notice of receipt of Application for Design Certification for System 80+ Standardized Nuclear Power Plant Design. FICHE: 58893 051 acn: 9105090143
May 1, 1991	D. Crutchfield, NRC, letter forwarding FR notice of receipt of CE March 30, August 21, 1989, April 26, July 12, October 29, 1990 and March 4, 1991 applications for CP and OL regarding System 80+ standardization nuclear power plant design. (Correction May 15, 1991) FICHE: 58893 047 acn: 9105090039
May 6, 1991	E.H. Kennedy, letter forwarding response to NRC January 31, 1991, request for additional information regarding CE standard SAR - design certification. FICHE: 57832 021 acn: 9105230164
May 13, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on CESSAR-DC System 80+ Chapter 15 within 90 days. FICHE: 57826 349 acn: 9105170260
May 13, 1991	A.E. Scherer, letter proposing series of meetings with NRC to identify critical path issues and explore areas for schedule improvements. FICHE: 57897 283 acn: 9105300069
May 13, 1991	E.H. Kennedy, letter responding to NRC April 12, 1991, request for list of documentation regarding System 80+ design certification application. FICHE: 57891 177 acn: 9105290234
May 15, 1991	C.L. Miller, correction to notice of receipt of application for design certification. FICHE: 57881 153 acn: 9105220272
May 15, 1991	T.V. Wambach, NRC, letter forwarding correction to Notice of Receipt of Application for Design Certification. FICHE: 57881 152 acn: 9105220266
May 16, 1991	E.H. Kennedy, letter responding to NRC request for additional information regarding review of CESSAR. FICHE: 57871 001 acn: 9105300020
May 22, 1991	T.J. Kenyon, NRC, meeting summary of May 14, 1991 meeting with NUMARC, EPRI, GE, CE, and Westinghouse regarding schedules for review of future LWR projects. FICHE: 57917 279 acn: 9106040123

June 13, 1991	Text-safety report - "Description of Nuclear QA Program." FICHE: 58237 009 acn: 9106260402
June 13, 1991	E.H. Kennedy, letter forwarding Revision 6 to topical report CENPD-210 "Description of Nuclear QA Program." FICHE: 58237 001 acn: 9106260398
June 26, 1991	E.H. Kennedy, letter advising that "System 80+ Standard Design Probabilistic Risk Assessment" to be made available to NRC. FICHE: 58279 316 acn: 9107020329
July 2, 1991	T.V. Wambach, NRC, letter notifying of July 16, 1991, presentation in Rockville, Maryland to discuss overview of CE System 80+ design and new features. FICHE: 58456 213 acn: 9107180028
July 22, 1991	D. Crutchfield, NRC, letter requesting list of assumptions used to develop estimated schedule for certification of System 80+ reactor design by November 4, 1991. FICHE: 58549 027 acn: 9107240322
July 29, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on Electrical System Branch of Chapter 8 of CESSAR-DC System 80+ standard design documents. FICHE: 58637 320 acn: 9108010176
August 2, 1991	C.L. Miller, NRC, letter requesting submittal of Revision 0 to DCTR-RS-02 "System 80+ Standard Design PRA" for evaluation in support of application of design certification for System 80+. FICHE: 58723 076 acn: 9108090123
August 3, 1991	T.V. Wambach, NRC, letter requesting additional information on CESSAR-DC System 80+ based on review by Radiation Protection Branch of Chapters 9, 11 12, and 15. FICHE: 58708 212 acn: 9108080049
August 5, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review by Reactor Safeguards Branch of CESSAR-DC. Response requested within 90 days of letter receipt. FICHE: 58717 289 acn: 9108080083
August 6, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review by Performance and Quality Evaluation Branch of CESSAR-DC. Response needed within 90 days of receipt of request. FICHE: 58722 265 acn: 9108090200

August 8, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review by Materials and Chemical Engineering Branch of CESSAR-DC. Response needed within 90 days of receipt of request. FICHE: 58774 102 acn: 9108140086
August 9, 1991	E.H. Kennedy, CE, letter advising that Revision 0 to DCTR-RS-02 "System 80+ Standard Design PRA Documentation" need not be protected from public disclosure. Report submitted for NRC information only. FICHE: 58822 321 acn: 9108190013
August 9, 1991	E.H. Kennedy, CE, letter advising that licensee will not be able to respond to request for additional information by requested date and assessing schedule for completion. FICHE: 58805 195 acn: 9108160067
August 14, 1991	Text-safety report - Volumes 1 through 3 of "System 80+ Standard Design PRA." FICHE: 58809 121 acn: 9108150250
August 14, 1991	E.H. Kennedy, letter forwarding Volumes 1 and 2 of "System 80+ Standard Design PRA." FICHE: 58809 120 acn: 9108150248
August 14, 1991	T.V. Wambach, NRC, meeting summary of June 20 and 21, 1991, with BNL and DC in Windsor, Connecticut to provide introduction to NRC reviewers for System 80+ PRA. List of attendees and viewgraphs enclosed. FICHE: 58897 167 acn: 9108260044
August 21, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on DC application for ABB-CE System 80+ design CESSAR-DC. FICHE: 58898 037 acn: 9108270128
September 4, 1991	Transcript of ACRS Subcommittee on Advanced PWRS Proceedings on September 4, 1991 in Bethesda, Maryland to receive presentation from ABB-CE regarding System 80+ Nuplex Advanced Instrumentation and Control Systems and PRA applied to design. PP 1-272. Viewgraphs enclosed. FICHE: 59075 001 acn: 9109100095
September 16, 1991	D. Crutchfield, NRC, letter requesting response to outstanding CESSAR-DC submittals regarding design certification of System 80+ within 120 days of August 9, 1991, letter. Listed areas included: interface requirements and fire hazards analysis. FICHE: 59186 335 acn: 9109230044
September 19, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on CESSAR-DC System 80+. Requests response within 90 days of receipt of September 19, 1991 letter. FICHE: 59288 038 acn: 9110030280

September 25, 1991	T.V. Wambach, NRC, letter requesting additional information regarding CESSAR-DC System 80+ based on audits by NRC. FICHE: 59288 158 acn: 9110030179
September 26, 1991	T.V. Wambach, NRC, letter forwarding request for additional information regarding CESSAR-DC System 80+. FICHE: 59294 110 acn: 9110040245
October 9, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review of CESSAR-DC regarding emergency planning. FICHE: 59408 189 acn: 9110180136
October 10, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review by Plant Systems Branch of Chapters 3, 5, 6, 9, 11, and Chapter 20. FICHE: 59479 293 acn: 9110300084
October 10, 1991	T.V. Wambach, NRC, letter forwarding request for additional information on CESSAR-DC System 80+ in reference to Generic Safety Issue II.C.4, "Reliability Engineering," to enable continuation of review of subject generic issue. FICHE: 59437 222 acn: 9110230078
October 10, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review of CESSAR-DC Chapter 14.2 regarding review evaluation and approval of Phases I through IV test results. FICHE: 59408 050 acn: 9110180137
October 16, 1991	T.V. Wambach, NRC, letter forwarding request for additional information based on review by TSS Branch of Chapter 16 of CESSAR-DC/System 80+. Requests response be provided within 90 days. FICHE: 59433 023 acn: 9110210196
October 23, 1991	D. Crutchfield, NRC, forwards letter revised agenda for NRC advanced conference to be held in Washington, DC on November 4 and 5, 1991. Agenda based on comments received from NUMARC on preliminary agenda in FR. List of technical issues for CE System 80+ enclosed. FICHE: 59491 306 acn: 9110310124
October 24, 1991	T.V. Wambach, NRC, meeting summary of August 12, 1991, meeting with licensee in Rockville, Maryland regarding licensee April 12, 1991, response to NRC request for additional information regarding human factors engineering for CE System 80+ dated December 21, 1990. FICHE: 59488 153 acn: 9110310083

October 30, 1991	T.V. Wambach, NRC, forwarding letter submitting request for additional information regarding review by probabilistic risk assessment branch of Appendix B. Response request within 90 days. FICHE: 59897 162 acn: 9112040040
November 5, 1991	D. Crutchfield, NRC, forwards draft safety evaluation regarding review of application certification of advanced BWR design. FICHE: 60519 178 acn: 9202100117
November 14, 1991	D. Crutchfield, NRC, forwards letter providing comments regarding E.H. Kennedy October 22, 1991, letter to D.M. Crutchfield concerning NRC request for schedule for outstanding submittals. Encourages licensee to meet or beat dates indicated with high quality submittals regarding review of CE System 80+. FICHE: 59880 031 acn: 9112040036
November 21, 1991	D. Crutchfield, NRC, forwarding letter requesting submittal of design certification to assess severe accident mitigation design alternatives and impact on safety of all designs. FICHE: 59912 283 acn: 9112060216
November 27, 1991	E.H. Kennedy, CE, forwarding letter response to NRC May 13 and August 21, 1991, requests for additional information to enable NRC to continue review of CESSAR for design certification. Topics discussed include definition of single failure and sequence of events. FICHE: 59979 017 acn: 9112110118
November 27, 1991	E.H. Kennedy, CE, forwarding letter nonproprietary and proprietary System 80+ seismic response spectra for soil Case B4 and System 80+ SSE Analysis Zero period accelerations. With 11 oversize drawings. FICHE: 60307 113 acn: 9201070320
November 27, 1991	E.H. Kennedy, CE, forwarding letter response to NRC August 8, 1991, request for additional information on CESSAR design certification. FICHE: 59957 193 acn: 9112100107
November 27, 1991	E.H. Kennedy, CE, forwarding letter response to NRC August 3, 1991, request for additional information to enable NRC to continue review of engineering standard SAR for design certification. FICHE: 59957 238 acn: 9112100045
November 30, 1991	Text-safety report - Nonproprietary "System 80+ SSE Analysis Zero Period Accelerations" including technical information on stability (Non-Buckling) calculation. With 11 oversize drawings. FICHE: 60307 123 acn: 9201070336

November 30, 1991	Text-safety report - Nonproprietary "System 80+ Seismic Response Spectra for Soil Case B4." FICHE: 60307 120 acn: 9201070332
November 30, 1991	CE, Proprietary "System 80+ Seismic Response Spectra for Soil Case B4." FICHE: 98214 265 acn: 9201070341
November 30, 1991	CE, Proprietary "System 80+ SSE analysis, Zero Period Accelerations. FICHE: 98214 268 acn: 9201070345
December 2, 1991	E.H. Kennedy, CE, forwarding letter response to NRC July 22, 1991, request that vendor provide information on basis for System 80+ review schedule. Schedules discussed in May 1991 no longer considered realistic. Listed projected dates discussed during advanced LWR Conference on November 4 and 5, 1991. FICHE: 59979 002 acn: 9112110013
December 9, 1991	E.H. Kennedy, CE, forwarding letter nonproprietary and proprietary reports regarding functional design requirement for CEA calculator and functional design requirement for core protection calculator respectively. Proprietary reports withheld (Ref 10 CFR 2.790). FICHE: 60058 220 acn: 9112190175
December 13, 1991	E.H. Kennedy, CE, forwarding letter response to request for additional information regarding standard SAR design certification. FICHE: 60102 302 acn: 9112260251
December 17, 1991	E.H. Kennedy, CE, forwarding letter response to NRC August 5, 1991, request for additional information on CESSAR design certification. FICHE: 60139 057 acn: 9112260243
December 20, 1991	T.V. Wambach, NRC, meeting summary of November 20, 1991, meeting with CE utility and EPRI in Rockville, Maryland regarding electrical distribution system modifications. FICHE: 60138 226 acn: 9112300312
December 20, 1991	E.H. Kennedy, CE, letter forwarding response to NRC July 29, 1991, request for additional information regarding Engineering SSAR - design certification covering IEEE Standards Offsite Power System Control Room indication of emergency diesel generator, operational status, and dc power system. FICHE: 60240 279 acn: 9201080036
December 21, 1991	Text-safety report - Amendment I to "CESSAR - Design Certification (CESSAR-DC)." FICHE: 57013 031 acn: 9103130321

December 23, 1991	E.H. Kennedy, CE, letter responding to NRC November 21, 1991, letter which summarized basis for considering severe accident mitigation design alternatives a part of design certification reviews. Alternative cost benefit analysis will be submitted by May 1992. FICHE: 60213 073 acn: 9201060010
December 23, 1991	T.V. Wambach, NRC, meeting summary of November 22, 1991 meeting with CE in Rockville, Maryland regarding use of design acceptance criteria for instrumentation and control and human factors. FICHE: 60163 310 acn: 9112310222
December 24, 1991	E.H. Kennedy, CE, letter responsing to NRC August 3, 1991, request for additional information regarding SSAR design certification. FICHE: 60238 272 acn: 9201070296
December 24, 1991	E.H. Kennedy, CE, letter responding to NRC requests for additional information for review of CESSAR - design certification report. Proposed revisions to SSAR enclosed. FICHE: 60238 140 acn: 9201070251
January 1, 1992	E.H. Kennedy, CE, letter forwarding response to NRC requests for additional information from Reactor Systems, Plant Systems, and Risk Assessment Branches regarding shutdown risk. FICHE: 60471 034 acn: 9202030204
January 6, 1992	R.C. Pierson, NRC, forwarding Generic Letter 82-39 "Problems with Submittals for 10 CFR 73.21." Requests that all future submittals containing safeguards information adhere to guidance contained in generic letter. FICHE: 60250 267 acn: 9201100038
January 8, 1992	T.V. Wambach, NRC, letter forwarding draft NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation." Final version of NUREG scheduled to be issued to Commission by early February 1992. FICHE: 60384 157 acn: 9201270311
January 9, 1992	T.V. Wambach, NRC, memorandum notifying January 21, 1992, meeting with CE in Windsor, Connecticut to discuss status of System 80+ and identification of significant issues in need of prioritization. Meeting will be extended through January 22, 1992, if necessary. FICHE: 60321 104 acn: 9201160163
January 12, 1992	E.H. Kennedy, CE, letter discussing E.H. Kennedy, January 10, 1992, response to NRC questions on 80 electrical distribution system. Supports EPRI effort to reach agreement with NRC on cost-effective design that meets needs of utility and NRC requirements. FICHE: 61666 341 acn: 9205110189

January 14, 1992	E.H. Kennedy, CE, letter forwarding response to NRC July 29, 1991, request for additional information for NRC review of CESSAR design certification. FICHE: 60339 202 acn: 9201210255
January 21, 1992	E.H. Kennedy, CE, letter discussing January 10, 1992, response to NRC questions on System 80+ electrical distribution system. FICHE: 61666 341 acn: 9205110189
January 23, 1992	E.H. Kennedy, CE, letter forwarding response to NRC October 30, 1991, request for additional information regarding CE standard SAR - design certification (CESSAR-DC). Responses to remaining questions will be provided by separate correspondence. FICHE: 60418 108 acn: 9201290113
January 24, 1992	E.H. Kennedy, CE, letter forwarding response to NRC request on October 10, 1992, for staff review of combustion engineering standard safety analysis report and corresponding revisions to CESSAR-DC. With 26 oversize drawings. FICHE: 60522 001 acn: 9201310151
January 24, 1992	E.H. Kennedy, CE, letter forwarding response to Materials and Chemical Engineering Branch August 8, 1992, Structural and Geosciences Branch September 26, 1992, and Plant Systems Branch October 10, 1991, requests for additional information regarding SSAR - design certification. FICHE: 60419 056 acn: 9201300093
January 28, 1992	E.H. Kennedy, CE, letter informing of revision to System 80+ general arrangements to reflect design improvements. With 13 oversize general arrangement drawings, 8 oversize fire barrier drawings, and 16 oversize radiation zone drawings. FICHE: 60449 317 acn: 9202030041
January 29, 1992	E.H. Kennedy, CE, letter forwarding responses to NRC requests for additional information regarding CE Standard SAR - design certification (CESSAR-DC) and marked-up proposed revisions to subject report. FICHE: 60471 127 acn: 9202040200
January 30, 1992	T.V. Wambach, NRC, letter informing that November 27, 1992, submittal of documents "System 80+ SSE Analysis Zero Period Accelerations" and "System 80+ Seismic Response Spectra for Soil Case B4" will be withheld from public disclosure referencing 10 CFR 2.790. FICHE: 60494 241 acn: 9202060176
January 31, 1992	E.H. Kennedy, CE, forwards letter "LOCA Aspects of CE Advanced LWR - System 80+" in response to NRC request for additional information. FICHE: 60570 019 acn: 9202110334

January 31, 1992	Text-specifications and test reports - draft Revision 0 to "Reliability Assurance Program Plan for System 80+ Nuclear Power Plant." FICHE: 60584 015 acn: 9202110169
January 31, 1992	E.H. Kennedy, CE, letter forwarding draft Revision 80+ to "Reliability Assurance Program Plan for System 80+ Nuclear Power Plant." FICHE: 60584 011 acn: 9202110166
February 3, 1992	T.V. Wambach, NRC, meeting summary of December 4, 1991, meeting with licensee in Rockville, Maryland regarding facility human factors issues. FICHE: 60540 089 acn: 9202110513
February 6, 1992	T.V. Wambach, NRC, letter advising that Revisions 02-P to CEN-304-P "Functional Design Requirement for Control Element Assembly Calculator" and CEN-305-P "Functional Design Requirements for Core Protection Calculator" withheld (Ref. 10 CFR 2.790). FICHE: 60538 254 acn: 9202110410
February 7, 1992	R.L. Nease, NRC, forwards meeting summaries-internal (non-transcript) summary of January 23, 1992, meeting with GE in Bethesda, Maryland to discuss closure of severe accident issues for Advanced BWR. FICHE: 60581 242 acn: 9202140363
February 11, 1992	S. Rosen, forwards text-safety report "LOCA Aspects of CE Advanced LWR - System 80+." FICHE: 60570 022 acn: 9202110335
February 12, 1992	C.B. Brinkman, CE, letter forwarding response to NRC October 10, 1991, letter regarding additional information for NRC staff review of CESSAR design certification. FICHE: 60750 055 acn: 9202260038
February 12, 1992	C.B. Brinkman, CE, letter forwarding response to NRC September 26, 1992, letter regarding additional information for review of CE standard SAR. With 2 oversize figure drawings. FICHE: 60773 128 acn: 9202250239
February 14, 1992	C.B. Brinkman, CE, letter responding to request for additional information regarding CE standard SAR design certification. FICHE: 60751 033 acn: 9202260092
February 18, 1992	C.B. Brinkman, CE, letter forwarding response to NRC August 6 and October 10, 1991, requests for additional information in reference to review of CESSAR design certification including corresponding revisions. FICHE: 60775 001 acn: 9202260175

February 18, 1992	C.B. Brinkman, CE, letter forwarding response to NRC October 30, 1991, request for additional information in reference to CESSAR design certification including corresponding revisions. With 2 oversize drawings. FICHE: 60774 180 acn: 9202260161
February 18, 1992	C.B. Brinkman, CE, letter forwarding response to NRC request for additional information electrical distribution system design in reference to NRC review of CESSAR. FICHE: 60750 183 acn: 9202260148
February 18, 1992	C.B. Brinkman, CE, letter forwarding NRC October 19, 1991, letter regarding additional information for NRC staff review of CESSAR design certification. FICHE: 60752 289 acn: 9202260141
February 19, 1992	C.B. Brinkman, CE, letter forwarding nonproprietary and proprietary responses to request for additional information including listing of MAAP parameter file for System 80+ design. Proprietary response and listing of MAAP parameter file for System 80+ design withheld. FICHE: 60751 328 acn: 9202260154
February 21, 1992	C.B. Brinkman, CE, letter forwarding "Description of Human Factors Program for the System 80+ (TM) Standard Plant Design" per commitment in December 4, 1991, meeting with NRC. FICHE: 60392 305 acn: 9203060317
February 25, 1992	C.B. Brinkman, CE, letter forwarding revised responses per December 4, 1991, meeting with NRC staff on human engineering factors on RAIs 620.5, 620.13, 620.16, 620.24, and 620.25. Other responses to be provided. FICHE: 60867 147 acn: 9203060293
February 25, 1992	C.B. Brinkman, CE, letter forwarding response to NRC September 26, 1992, request for additional information to enable NRC to continue review of SSAR - design certification (CESSAR-DC). Information covers damping values and groundwater condition. FICHE: 60858 074 acn: 9203060343
February 26, 1992	N.T. Saltos, letter discussing partial list of responses to request for additional information regarding plant. Without enclosure. FICHE: 60772 024 acn: 9203020274
February 28, 1992	C.B. Brinkman, CE, letter providing revised responses to NRC requests for additional information regarding human factors engineering per April 12, 1992, letter. FICHE: 60883 008 acn: 9203090118
February 28, 1992	C.B. Brinkman, CE, letter forwarding summary of interface requirements for System 80+ standard design and corresponding revisions to CESSAR - design certification. FICHE: 60883 076 acn: 9203090104

February 29, 1992	Text-specifications and test reports - "Description of Human Factors Program Plan for System 80+ (TM) Standard Plant Design" for plant certification. FICHE: 60392 307 acn: 9203060327
March 2, 1992	C.M. Thompson, letter forwarding trip report of November 4 through 6, 1991, visit to plant to identify areas where changes could be made to reduce radiation exposure and to incorporate changes into design review of advanced CESSAR System 80+ design. FICHE: 60907 297 acn: 9203120343
March 4, 1992	D. Crutchfield, NRC, letter requesting completion of responses to requests for additional information and submittal of remaining portions of application. FICHE: 60839 110 acn: 9203060134
March 4, 1992	T.V. Wambach, NRC, meeting summary of January 21, 1992, meeting with CE regarding CE goals and objectives with regard to System 80+, overview of System 80+, and major issues that require manpower resources to complete NRC evaluation. FICHE: 60912 172 acn: 9203130049
March 11, 1992	Text-safety report - general external technical reports, Section 17.3 regarding responses to resolution issues related to reliability assurance program. FICHE: 60898 284 acn: 9203110138
March 12, 1992	Text-safety report - markup copy of "QA Program Topical Report." FICHE: 60993 203 acn: 9203190036
March 12, 1992	Text-safety report - "QA Program Topical Report." FICHE: 60993 160 acn: 9203190033
March 12, 1992	C.B. Brinkman, CE, letter forwarding Revision 7 to CENPD-210, "QA Program Topical Report" and marked-up copy of Revision 6 to CENPD-210A, "QA Program Topical Report." FICHE: 60993 159 acn: 9203190031
March 13, 1992	Text-safety report - Revision 0 to "System 80+ Design Certification Fire Hazards Assessment." FICHE: 61222 217 acn: 9204010227
March 16, 1992	T.V. Wambach, NRC, meeting summary of March 2, 1992, with CE in Windsor, Connecticut regarding System 80+ design certification for instrumentation and control. Meeting agenda and list of attendees enclosed. FICHE: 60997 319 acn: 9203200177

March 18, 1992	T.V. Wambach, NRC, meeting summary of February 26, 1992, public meeting with CE in Rockville, Maryland regarding piping ductwork and cable trays concerning level of detail and CE distribution system guide as design acceptance criteria. FICHE: 61121 020 acn: 9203270296
March 19, 1992	General external technical reports - draft "Interim Human Factors Review Criteria for Design Process of Advanced Nuclear Power Reactor." FICHE: 61713 049 acn: 9205140246
March 25, 1992	C.B. Brinkman, CE, letter forwarding acknowledge receipt of March 4, 1992, letter regarding CESSAR-DC completeness of submittals. Agrees that licensee must continue to provide timely high quality submittals. Submittal target dates for shutdown risk information advanced from April 30 through May 31, 1992. FICHE: 61297 185 acn: 9204080186
March 26, 1992	C.B. Brinkman, CE, letter forwarding Revision 0 to "System 80+ Design Certification Fire Hazards Assessment." Assessment covered all areas containing equipment required for safe shutdown following fire. FICHE: 61222 216 acn: 9204010224
March 26, 1992	C.B. Brinkman, CE, letter discussing applicability of leak-before-break methodology in System 80+ design process per February 26, 1992, meeting with NRC. Walls of every System 80+ subcompartment sufficiently thick to withstand pressure effects from postulated guillotine rupture. FICHE: 61210 358 acn: 9204010016
March 26, 1992	C.B. Brinkman, CE, forwards letter response to remaining outstanding request for additional information (RAI) 230.1 and revisions to responses for RAIs 220.6 and 220.7 providing Figure 3.7-2, and expanding description of how average power spectral density calculated respectively. FICHE: 61231 298 acn: 9204020043
March 27, 1992	C.B. Brinkman, CE, letter submitting listing of deviations between acceptance criteria of NRC standard review plan and System 80+ design certification application, and submits standard review plan comments. FICHE: 61238 134 acn: 9204020200
April 3, 1992	D.M. Crutchfield, NRC, letter advising that NRC unable to confirm what CE proposing to certify under 10 CFR Part 52 for design of control room. FICHE: 71246 322 acn: 9204090057
April 7, 1992	C.B. Brinkman, CE, letter forwarding general arrangement fire barrier and radiation zone drawings in fulfillment of commitments of Items 1 and 2 of March 25, 1992, letter regarding CESSAR-DC System 80+. With 37 drawings. FICHE: 61352 324 acn: 9204150084

April 8, 1992	N.T. Saltos, letter forwarding document entitled "Fire Hazard Assessment" regarding CE System 80+ review. CE will prepare and submit PRA that will analyze potential of core damage from fires. Without enclosure. FICHE: 61378 294 acn: 9204210273
April 9, 1992	T.V. Wambach, NRC, letter forwarding request for additional information and description of SER regarding severe accident and design features for prevention and mitigation. Requests information in time frame to enable staff to meet schedule for draft SER. FICHE: 61378 129 acn: 9204210363
April 9, 1992	T.V. Wambach, NRC, letter forwarding request for information and description for SER to assist in closure of severe accident design features issues. FICHE: 71486 108 acn: 9209180094
April 15, 1992	C.B. Brinkman, CE, letter transmitting CESSAR-DC flow diagram matrix for auxiliary system. Enclosed tables specify CESSAR-DC figure correspondence to each drawing and whether each figure is new or replacement. FICHE: 61425 280 acn: 9204210238
April 15, 1992	C.B. Brinkman, CE, letter submitting Section 6.2.5 of design certification discussing layout of hydrogen mitigation system igniter locations. FICHE: 61422 315 acn: 9204210237
April 15, 1992	Text-safety report - "Selection of Control Motion for ABB-CE System 80+ Standard Design." FICHE: 61424 197 acn: 9204210234
April 15, 1992	C.B. Brinkman, CE, forwards letter, "Selection of Control Motion for ABB-CE System 80+ Standard Design." Report identifies and discussed hypothetical rock outcrop spectrum and control motions. FICHE: 61424 195 acn: 9204210232
April 15, 1992	T.V. Wambach, NRC, meeting summary of March 17, 1992, meeting with CE in Rockville, Maryland regarding Chapter 15 of plant accident analyses. FICHE: 61416 166 acn: 9204220142
April 21, 1992	J. O'Hara contracted report draft, "Interim Human Factors Review Criteria for Design Process of Advanced Nuclear Power Reactor." FICHE: 61822 287 acn: 9205280258
April 22, 1992	T.V. Wambach, NRC, memorandum notification of April 27 to 30, 1992, meetings with CE in Bethesda, Maryland to discuss geoscience and structural design audit, and potential open issues for draft SER. Agenda enclosed. FICHE: 61462 049 acn: 9204270246

April 22, 1992	T.V. Wambach, NRC, memorandum notification of April 24, 1992, meeting with CE in Rockville, Maryland to discuss System 80+ (control room human factors) review status. FICHE: 61462 069 acn: 9204270233
April 23, 1992	T.G. Hiltz, NRC, meeting summary of April 16 and 17, 1992, with CE regarding propose audit of System 80+ control room design content of draft inspection test analysis, attendance criteria and CE responses to NRC request for additional information regarding Chapter 18 of CESSAR-DC. List of attendees enclosed. FICHE: 61471 058 acn: 9204280331
April 23, 1992	T.V. Wambach, NRC, memorandum notification of April 30, 1992, meeting with CE in Bethesda, Maryland to present revised proposal for offsite power distribution design. FICHE: 61471 055 acn: 9204280317
April 24, 1992	C.B. Brinkman, CE, letter forwarding Revision 0 to "Design Alternatives for System 80+ Nuclear Power Plant" per commitment in March 25, 1992, letter. FICHE: 61552 001 acn: 9205010195
April 30, 1992	R.A. Matzie, CE, letter forwarding definition of Nuplex 80+ design to enable NRC to complete review of human factors for System 80+ and DCRDR audit per April 9, 1992, meeting. Table listing docketed documents enclosed. FICHE: 61650 291 acn: 9205110177
April 30, 1992	C.B. Brinkman, CE, letter forwarding nonproprietary and proprietary slides presented at March 19, 1992, meeting with NRC regarding System 80 reactor coolant pump seal design and performance as supplement to RAI responses in reference to March 25, 1992, letter. Proprietary slides withheld (Ref. 10 CFR 2.790). FICHE: 61633 012 acn: 9205070134
April 30, 1992	Text-safety report - draft "System 80+ Shutdown Risk Evaluation Report" Part 1. FICHE: 61591 168 acn: 9205070106
April 30, 1992	C.B. Brinkman, CE, letter forwarding draft "System 80+ Shutdown Risk Evaluation Report" Part 1 in fulfillment of commitment of Item 5 of March 25, 1992, letter. Part 2 of draft and final version will be provided later this summer. FICHE: 61591 167 acn: 9205070104
April 30, 1992	T.V. Wambach, NRC, meeting summary of April 9, 1992 in Rockville, Maryland to discuss scope of design certification for Nuplex 80+ and control room for System 80+. List of attendees enclosed. FICHE: 61602 294 acn: 9205070073

April 30, 1992	C.B. Brinkman, CE, letter forwarding draft Section 7 of distribution system design guide discussed at Mechanical and Piping System audit meeting on April 22 and 23, 1992, in Charlotte, North Carolina per commitment in vendor March 25, 1992 letter. FICHE: 61619 001 acn: 9205070039
April 30, 1992	C.B. Brinkman, CE, letter forwarding information to supplement March 25, 1992, responses to NRC request for additional information regarding analysis of ruptured reactor coolant pump, seal cooler tube, and single failure assumptions for analysis of moderate-energy-line break. FICHE: 61600 141 acn: 9205070024
April 30, 1992	C.B. Brinkman, CE, letter forwarding draft System 80+ Tier 1 descriptions, inspection tests analyses, and acceptance criteria for System 80+ standard design. FICHE: 61631 165 acn: 9205070020
April 30, 1992	C.B. Brinkman, CE, letter forwarding changes to Revision 7 to Topical Report CENPD-210 initially submitted to NRC via March 12, 1992, letter. Changes cover QA program, QA policy and responsibilities of QA organization. FICHE: 61550 115 acn: 9205040204
April 30, 1992	Text-safety report - Revision 0 to "Design Alternatives for System 80+ Nuclear Power Plant." FICHE: 61552 002 acn: 9205010196
May 1, 1992	T.V. Wambach, NRC, meeting summary of March 19, 1992, with CE, Inc., in Rockville, Maryland regarding integrity of RCP seals upon loss of coolant and intersystem LOCA considerations. List of attendees enclosed. FICHE: 61592 261 acn: 9205070244
May 6, 1992	R.L. Palla, NRC, letter submitting response from CE regarding System 80+ PRA. Information provided for review under Task Order 2 to FIN L-2412. FICHE: 62166 107 acn: 9206300160
May 6, 1992	T.V. Wambach, NRC, letter forwarding discussion paper to form basis for telcon or meeting to discuss open issues in staff review of reliability assurance program plan for System 80+. Requests to be informed of when applicant will be able to support telcon. FICHE: 61718 171 acn: 9205190283
May 7, 1992	L. Greimann, AMES LAB, letter forwarding input to draft SER for System 80+. FICHE: 71295 348 acn: 9205150031

May 8, 1992	C.B. Brinkman, CE, letter forwarding information to supplement previous RAI responses regarding System 80+. Enclosures include information regarding size and type of modeling elements used in analysis of containment structure and commitment to use more conservative radiological dispersion factors. FICHE: 61741 032 acn: 9205190154
May 8, 1992	C.B. Brinkman, CE, letter forwarding proprietary Revision 1 to NPX80-IC-SD790-02 "System Description for Critical Function and Success Path Monitoring in Nuplex 80+" and "System Description for Control Complex information for Nuplex 80+." Reports withheld. FICHE: 61717 306 acn: 9205180324
May 8, 1992	Text-safety report - "Nuplex 80+ Human Factors Design Process Summary." FICHE: 61781 098 acn: 9205220008
May 8, 1992	C.B. Brinkman, CE, letter forwarding "System 80+ Function and Task Analysis Final Report" and "Nuplex 80+ Revision 2 to Verification Analysis Report." FICHE: 61781 001 acn: 9205210189
May 14, 1992	Text-specifications and test reports - Revision 0 to "System 80+ Reactor Coolant Pump Seal Loss of Seal Cooling Test Data Report." FICHE: 61780 241 acn: 9205220144
May 14, 1992	C.B. Brinkman, CE, letter forwarding Revision 0 to DCTR 12, "System 80+ Reactor Coolant Pump Seal Loss of Seal Cooling Test Data Report" to fulfill commitment made at April 22 and 23, 1992, piping and mechanical design review meeting. FICHE: 61780 239 acn: 9205220133
May 17, 1992	C.B. Brinkman, CE, letter notifying of staff plans to incorporate all information docketed by CE by May 8, 1992, in draft SER for System 80+. FICHE: 71270 253 acn: 9204300398
May 18, 1992	C.B. Brinkman, CE, letter responding to issue of diversity for digital instrumentation and control system for System 80+ delineated in NRC April 30, 1992, letter. Best estimate analysis underway in order to make realistic assessment of plant performance given computer failure. FICHE: 61884 189 acn: 9205280060
May 19, 1992	T.H. Boyce, NRC, memorandum notification of May 20, 1992, meeting with CE for System 80+ in Rockville, Maryland to discuss initial comments on CE pilot ITAAC submittal. FICHE: 61756 349 acn: 9205210314

May 21, 1992	R.C. Pierson, NRC, letter forwarding initial comments on pilot Tier 1 design information, inspection tests analyses, and acceptance criteria (ITAAC) submittal for System 80+ per April 30, 1992, submittal. NRC should be advised of schedule for complete Tier 1 and ITAAC submittal. FICHE: 61822 063 acn: 9206010196
May 22, 1992	C.B. Brinkman, CE, letter forwarding report NPLEX-IC-DR-791-02 "Human Factors Engineering Standards Guidelines and Bases for Nuplex 80+." Report provides information to supplement previous RAI responses on human factors engineering of control room in fulfillment of April 17, 1992, commitments. FICHE: 61904 028 acn: 9206020175
May 22, 1992	C.B. Brinkman, CE, letter discussing System 80+ human factors engineering review criteria per May 19, 1992, meeting. Text of draft SER should clearly state that work continuing with ABB-CE through public meetings to converge on set of criteria for review of Nuplex 80+. FICHE: 61922 266 acn: 9206020003
May 26, 1992	R.A. Matzie, CE, letter requesting mailing lists for System 80+ project be revised as indicated on enclosure. E.H. Kennedy should be replaced with C.B. Brinkman. FICHE: 61922 258 acn: 9206020006
May 28, 1992	T.V. Wambach, NRC, meeting summary of April 22 and 23, 1992, public meetings with ABB-CE and Duke Energy and Services in Charlotte, North Carolina regarding mechanical design criteria for CE System 80+. List of attendees, agenda and viewgraphs enclosed. FICHE: 61906 243 acn: 9206040214
May 28, 1992	T.V. Wambach, NRC, meeting summary of April 30, 1992, public meeting with ABB-CE regarding modification to System 80+ electrical distribution system. List of attendees and presentation material enclosed. FICHE: 61921 287 acn: 9206020302
May 29, 1992	D. Gallagher, SAIC, letter forwarding draft "Technical Evaluation Report for Containment System of CESSAR System 80+." FICHE: 71367 247 acn: 9206250239
May 29, 1992	D.M. Crutchfield, NRC, letter forwarding announcement and invitation for fourth annual NRC regulatory information conference to discuss status of CE System 80+ licensing effort since November 1991 ALWR conference. FICHE: 61882 197 acn: 9206030360
May 29, 1992	C.B. Brinkman, CE, letter forwarding description of human reliability analysis methodology used in probabilistic risk assessment. FICHE: 61997 243 acn: 9206110109

May 31, 1992	Text-safety report - "Human Factors Engineering Standards Guidelines and Bases for Nuplex 80+." FICHE: 61904 033 acn: 9206020177
June 3, 1992	T.V. Wambach, NRC, meeting summary (non-transcript) of April 27 to 30, 1992, public meeting in Bethesda, Maryland regarding seismic and structural design issues of System 80+. List of attendees, agenda, and material presented each day enclosed. FICHE: 62035 198 acn: 9206160259
June 3, 1992	T.V. Wambach, NRC, meeting summary of April 28, 1992, meeting with ABB-CE in Windsor, Connecticut to discuss PRA update and ABB-CE answers to RAIs. FICHE: 62021 064 acn: 9206120315
June 3, 1992	T.V. Wambach, NRC, meeting summary of April 24, 1992, meeting with ABB-CE regarding status and direction of NRC Human Factors engineering review of Nuplex 80+ Control Room for CE System 80+. List of Attendees enclosed. FICHE: 62021 001 acn: 9206120300
June 3, 1992	C.B. Brinkman, CE, letter forwarding marked-up draft "Defense Against Common Mode Failures in Digital Instrumentation and Control System" per June 1, 1992, meeting with NRC. FICHE: 62040 001 acn: 9206120219
June 3, 1992	T.V. Wambach, NRC, meeting summaries-internal (non-transcript) summary of April 24, 1992 meeting with ABB-CE regarding status and direction of NRC Human Factors Engineering Review of Nuplex 80+ Control Room for CE System 80+. List of attendees enclosed. FICHE: 62021 001 acn: 9206120300
June 4, 1992	R.C. Pierson, NRC, meeting summary of May 19, 1992, meeting with licensee in Windsor, Connecticut regarding Chapter 18 of CESSAR-DC (Human Factors). FICHE: 62032 228 acn: 9206150445
June 5, 1992	C.B. Brinkman, CE, letter advising that Tier 1 design information, inspections, tests, analyses, and acceptance criteria submittal for System 80+ will be provided by October 16, 1992. FICHE: 62155 029 acn: 9206290010
June 10, 1992	T.V. Wambach, NRC, meeting summary of June 1, 1992, public meeting regarding draft staff position on diversity of instrumentation and controls for CE System 80+ nuclear plant and forwards list of attendees and other discussed material FICHE: 62036 333 acn: 9206160162

June 15, 1992	C.B. Brinkman, CE, letter forwarding proprietary special report CE NPSD-741-P "Evaluation of Design Features Which Minimize Probability of Interfacing System LOCAS for System 80+ Standard Design." With 8 oversize drawings. Report and drawings withheld. FICHE: 62223 070 acn: 9206290152
June 15, 1992	C.B. Brinkman, CE, letter submitting supplemental information for previous System 80+ RAIs including response to TMI Item II.E.3.1 which confirms that backup pressurizer heaters are not relied upon or required for natural circulation in RCS. FICHE: 62110 237 acn: 9206240349
June 15, 1992	Text-safety report - "System 80+ Shutdown Risk Evaluation Report." FICHE: 62220 002 acn: 9206260099
June 16, 1992	C.B. Brinkman, CE, letter forwarding draft DCTR 10 "System 80+ Shutdown Risk Evaluation Report," in fulfillment of commitment from March 25, 1992, letter. Final version of shutdown risk report will be provided by July 31, 1992. With 8 oversize drawings. FICHE: 62220 001 acn: 9206260094
June 25, 1992	T.V. Wambach, NRC, letter informing that April 30, 1992, submittal containing proprietary information will be withheld from public disclosure per 10 CFR 2.790. FICHE: 62204 213 acn: 9207060051
July 7, 1992	C.B. Brinkman, CE, letter forwarding "Criteria for Design of Main Control Room and Other Operating Stations for System 80+" in fulfillment of May 22, 1992, commitment. FICHE: 62450 215 acn: 9207200021
July 8, 1992	T.V. Wambach, NRC, letter informing that proprietary attachments to February 19, 1992, submittal will be withheld from public disclosure (Ref. 10 CFR 2.790) per February 20, 1992, affidavit. FICHE: 62338 002 acn: 9207130179
July 25, 1992	Text-safety report - draft "System 80+ Design Certification Distribution System Design Guide." FICHE: 62708 002 acn: 9208070100
July 28, 1992	S. Dembek, NRC, meeting summary of July 9, 1992, public meeting in Windsor, Connecticut regarding Human Factors Engineering design issues. List of meeting attendees and ABB-CE presentations enclosed. FICHE: 62645 329 acn: 9208050091

July 31, 1992

C.B. Brinkman, CE, letter forwarding System 80+ Human Factors Engineering Team description and markup of Part II of Human Factors Criteria document submitted by July 7, 1992, letter per July 9, 1992, meeting with NRC.
FICHE: 62717 295
acn: 9208100117

July 31, 1992

C.B. Brinkman, CE, letter forwarding draft "System 80+ Design Certification Distribution System Design Guide" per March 25, 1992, commitment. Proposes to meet with NRC after corresponding analyses completed in August.
FICHE: 62708 001
acn: 9208070098

July 31, 1992

Text-safety report - general external technical reports final "System 80+ Shutdown Risk Evaluation Report."
acn: 9208070093

July 31, 1992

C.B. Brinkman, CE, letter forwarding final DCTR 10 "System 80+ Shutdown Risk Evaluation Report," as described in March 25, 1992, letter regarding submittal schedule update. With three oversize drawings.
FICHE: 62578 001
acn: 9208070087

July 31, 1992

Text-specifications and test reports - "System 80+ Probabilistic Risk Assessment Program Plan."
FICHE: 62706 337
acn: 9208070043

July 31, 1992

C.B. Brinkman, CE, letter forwarding "System 80+ Probabilistic Risk Assessment Program Plan" for revised PRA per March 25, 1992, commitment.
FICHE: 62706 336
acn: 9208070036

July 31, 1992

Text-safety report - "Criteria for Design of Main Control Room and Other Operating Stations for System 80+."
FICHE: 62450 216
acn: 9207200024

August 10, 1992

C.B. Brinkman, CE, letter forwarding draft revision of pilot ITAAC for System 80+ standard design in response to NRC May 21, 1992, comments on April 30, 1992, submittal.
FICHE: 62828 143
acn: 9208180181

August 19, 1992

C.B. Brinkman, CE, letter forwarding Amendment J to "CESSAR - Design Certification." Amendment includes revisions proposed in previous letter; editorial changes and revisions previously transmitted informally. Applicant for review of CESSAR - Design Certification also enclosed.
FICHE: 63010 001
acn: 9209030241

August 28, 1992

C.B. Brinkman, CE, letter forwarding responses to April 8, 1992, request for additional information, regarding System 80+ severe accident design features.
FICHE: 63149 001
acn: 9209140134

August 31, 1992	C.B. Brinkman, CE, forwarding engineering report detailing design of System 80+ distribution system (piping, HVAC duct work and electrical cable trays) and draft "System 80+ Design Certification Piping Analysis Specification." FICHE: 63211 001 acn: 9209160316
September 2, 1992	C.B. Brinkman, CE, letter forwarding reports on approach to assessment of flood and fire protection in System 80+ PRA, including "PRA Flood Protection Assessment" and "Fire Hazards Risk Assessment (Phase I)." FICHE: 63152 168 acn: 9209140176
September 16, 1992	C.B. Brinkman, CE, letter forwarding structural models for System 80+ seismic analysis for nuclear island and nuclear annex, per April 1992, audit meeting in Bethesda, Maryland. FICHE: 63334 125 acn: 9209300024
September 16, 1992	C.B. Brinkman, CE, letter advises that vendor in process of developing Tier 1 design descriptions and accompanying ITAAC for System 80+. FICHE: 63418 359 acn: 9210050007
September 23, 1992	C.B. Brinkman, CE, letter forwarding Revision O to NPX80-IC-DP-790-01, "Nuplex 80+ Advanced Control Complex Design Bases," to support human factors engineering review of System 80+ control room. FICHE: 63478 161 acn: 9210130368
September 24, 1992	D. Tang, NRC, provides review comments on draft Ames Lab report, "System 80+ Containment - Structural Design Review." FICHE: 71511 039 acn: 9210010177
September 29, 1992	C.B. Brinkman, CE, letter forwarding ALWR-IC-DCTR-31, "Evaluation of Defense-in-Depth and Diversity in ABB-CE Nuplex 80+ Advanced Control Complex for System 80+ Standard Design," in response to NRC request during March 2, 1992 meeting. FICHE: 63481 233 acn: 9210130177
October 1, 1992	R. Pierson, NRC, letter forwarding draft SER of staff review of CESSAR for design certification of System 80+ (NUREG-1462). FICHE: 63501 001 acn: 9210160224
October 16, 1992	C.B. Brinkman, CE, describes change in QA/QC alignment of ABB-CE Nuclear Fuel FICHE: 63684 280 acn: 9210260305
October 16, 1992	C.B. Brinkman, CE, letter forwarding Level 1 PRA revision of "System 80+ Std Design PRA," including Chapter 1 through 6, 9 and 10. FICHE: 63848 001 acn: 9211060162

October 16, 1992	C.B. Brinkman, CE, letter forwarding seismic analysis structural model details for System 80+, requested by NRC September 23, and October 17, 1992, telcons, consisting of schematic of nuclear island and nuclear annex structures and table listing soil layers and properties. FICHE: 63947 310 acn: 9211180032
October 19, 1992	R. Pierson, NRC, discusses staff position on use of revised source term for CE System 80+. Proposed application of revised reactor accident source term to evolutionary LWR would be conducted on case-by-case basis. FICHE: 63638 157 acn: 9210230307
October 19, 1992	R. Pierson, NRC, letter forwarding detailed comments on revised pilot inspections, tests, analyses, and acceptance criteria submittal for System 80+ per vendor August 10, 1992, submittal. FICHE: 63649 297 acn: 9210270230
November 18, 1992	C.B. Brinkman, CE, letter forwarding response to 56 issues identified in draft SER for System 80+, sorted by review branch and including Amendment K to CESSAR design certification. FICHE: 64060 015 acn: 9211250142
November 19, 1992	D. Crutchfield, NRC, discusses November 6, 1992, meeting w/ABB-Combustion Engineering, regarding human factors review of Nuplex 80+ CR design. FICHE: 64005 249 acn: 9211250226
November 24, 1992	C.B. Brinkman, CE, letter forwarding responses to 359 issues identified in DSER for System 80+. FICHE: 64242 001 acn: 9212160080
December 1, 1992	C.B. Brinkman, CE, letter forwarding draft material intended to provide specific examples of Tier 1 material and related information as basis to discuss ITAAC at December 9, 1992, with NRC. FICHE: 64089 324 acn: 9212040042
December 15, 1992	C.B. Brinkman, CE, letter forwarding proprietary oversize Systems 80+ drawings, D200-9, 11 and 13, Revision 1 and electronic diskette with proprietary design and operations data for MAAP computer code. FICHE: 64606 222 acn: 9212280076
December 18, 1992	C.B. Brinkman, CE letter advising that report, "Design Review of Inter-System LOCA," transmitted by June 15, 1992, letter is not proprietary and report number that appears in upper-right hand corner of each page should be crossed-out. FICHE: 64444 271 acn: 9212290164

December 18, 1992	C.B. Brinkman, CE, letter forwarding responses to 154 of issues identified in DSER for System 80+. FICHE: 64436 001 acn: 9212300112
December 21, 1992	C.B. Brinkman, CE, letter forwarding Amendment K to "CESSAR - Design Certification." FICHE: 64469 001 & 64482 029 acn: 9212290220 & 9301060018
December 22, 1992	T.V. Wambach, NRC, letter forwarding LLNL December 3, 1992, report entitled, "Review of CE System 80+ FMEA and D&DID Analysis," to be discussed at January 6, 1993, public meeting. FICHE: 64435 254 acn: 9212310010
December 23, 1992	C.B. Brinkman, CE, letter forwarding Section 17.3, "Design Reliability Assurance Program Plan for System 80+ Standard Design," Revision 1 and responses to 79 issues identified in DSER for System 80+. FICHE: 64463 059 acn: 9212300096
December 23, 1992	M.X. Franovich, NRC, letter forwarding request for additional information on CESSAR-DC, System 80+ shutdown risk evaluation report. FICHE: 64439 329 acn: 9301040133
January 7, 1993	R.C. Jones, NRC, letter forwarding requests additional information on Topical Report CENPD-382-P, "Methodology for Core Designs Containing Erbium Burnable Adsorbers." FICHE: 71599 206 acn: 9301130167
January 14, 1993	R. Pichumani, NRC, letter forwarding diskettes containing CARES source program and CE System 80+ model and documentation of model contained in diskette, per January 13, 1993, telecon. FICHE: 71615 360 acn: 9302010122
January 14, 1993	C.B. Brinkman, CE, letter forwarding response to NRC October 19, 1992, request for additional information regarding revised radiological source term for System 80+. FICHE: 64757 169 acn: 9302020402
January 15, 1993	R.W. Borchardt, NRC, discusses level of design detail for System 80+ structures and lists expected level of completion. FICHE: 64639 312 acn: 9301250091
January 18, 1993	C.B. Brinkman, CE, letter forwarding responses to 54 issues identified in draft SER for System 80+. FICHE: 64712 036 acn: 9301280195

January 20, 1993	C.B. Brinkman, CE, letter forwarding responses to 232 issues identified in draft SER for System 80+. FICHE: 64715 001 acn: 9301280171
January 26, 1993	C.B. Brinkman, CE, letter forwarding responses to 23 issues identified in draft SER for System 80+. FICHE: 64761 001 acn: 9302030009
January 28, 1993	C.B. Brinkman, CE, letter forwarding draft System 80+ certified design description and associated ITAAC. FICHE: 64842 199 acn: 9302120222
January 29, 1993	M.E. Waterman, NRC, letter forwarding Amendment K to System 80+ ALWR CESSAR-DC. FICHE: 74417 087 acn: 9303310202
February 1, 1993	C.B. Brinkman, CE, letter forwarding additional certified design descriptions and ITAAC for System 80+. FICHE: 64807 054 acn: 9302100207
February 2, 1993	C.B. Brinkman, CE, letter forwarding responses to 22 issues, regarding closure of System 80+ draft SER issues. FICHE: 64825 143 acn: 9302110165
February 9, 1993	C.B. Brinkman, CE, submits summary of CESSAR-DC safety analyses reanalyzed to reflect resolution of DSER open items. FICHE: 64926 175 acn: 9302190293
February 10, 1993	D. Crutchfield, NRC, discusses ITAAC submittal for ABB-CE System 80+ and provides proposed schedule for review. Fiche: 64816 008 acn: 9302120153
February 16, 1993	C.B. Brinkman, CE, letter forwarding "System 80+ Advanced LWR, PRA-Based Seismic Margin Evaluation" and draft UCRL-CR-11478, "Basis for Seismic Provisions of UCRL-15910," per January 4, 1993, meeting with NRC. FICHE: 74007 233 acn: 9302240271
February 24, 1993	C.B. Brinkman, CE, letter forwarding Volumes 1, 2 and 3 of Revision 1 to DCTR-RS-02, "System 80+ Std Design PRA. FICHE: 74164 001 acn: 9303040310
March 1, 1993	J.N. Wilson, NRC, forwards SECY-93-041, "Advanced BWR Review Schedule." FICHE: 74098 320 acn: 9303050113

March 2, 1993	C.B. Brinkman, CE, requests that NRC provide brief assessment of present System 80+ review status, as followup to NRC September 28, 1992, issuance of NUREG-1462 (draft SER for System 80+). FICHE: 74296 353 acn: 9303150198
March 3, 1993	C.B. Brinkman, CE, letter forwarding figures showing translational acceleration response spectra in three directions at 91.75 ft elevation of containment interior structure per January 5, 1993, meeting request. FICHE: 74260 356 acn: 9303150005
March 3, 1993	C.B. Brinkman, CE, letter forwarding responses to questions on shutdown risk delineated in NRC December 23, 1992, letter and corresponding revisions to shutdown risk report submitted in vendor July 31, 1992, letter. FICHE: 74259 001 acn: 9303150178
March 4, 1993	C.M. Trammell, NRC, expresses appreciation for opportunity to visit plant on February 9 and 10, 1993, to interview operators regarding operating experience with CE System 80+ design that would be pertinent to NRC review of CE System 80+ advanced control room design. FICHE: 74149 102 acn: 9303100128
March 4, 1993	C.B. Brinkman, CE, letter forwarding Revision 0 to NPX80IC-RR790-02, "Human Factors Evaluation and Allocation of System 80+ Functions." FICHE: 74286 134 acn: 9303150103
March 5, 1993	C.B. Brinkman, CE, letter forwarding information requested by NRC to supply information on System 80+ design description already on docket. FICHE: 74276 213 acn: 9303160019
March 5, 1993	C.B. Brinkman, CE, letter forwarding 11 prototype system design descriptions and associated ITAAC for review and approval. FICHE: 74298 250 acn: 9303160051
March 5, 1993	C.B. Brinkman, CE, transmits Amendment L to "CESSAR-Design Certification." FICHE: 74441 001 acn: 9303180124
March 5, 1993	S.A. Toelle, letter forwarding "Small Break LOCA Realistic Evaluation Model" Topical Report, CEN 420-P, Volume 1, Part 1, describing calculation models. FICHE: 74559 255 acn: 9304120127
March 10, 1993	C.B. Brinkman, CE, letter forwarding "ABB-CE System 80+ Progress Report on Structural Analysis of Nuclear Island and Nuclear Annex Structures," to support March 17 and 18, 1993, meeting in Charlotte, North Carolina consisting of std structural design criteria draft spec. FICHE: 74278 080 acn: 9303160101

March 10, 1993	C.B. Brinkman, CE, letter forwarding report on protection against common mode failure of digital instrumentation and control system as agreed at January 21, 1993, meeting with NRC. FICHE: 74279 153 acn: 9303160121
March 17, 1993	T.E. Murley, NRC, responds to March 4, 1993, letter, regarding SECY-93-041, "ABWR Review Schedule." FICHE: 74309 225 acn: 9303190207
March 17, 1993	C.B. Brinkman, CE, letter forwarding summary of how design and operating experience incorporated into System 80+ design process. FICHE: 74378 309 acn: 9303250153
March 17, 1993	C.B. Brinkman, CE, letter forwarding response to supplementary questions from Sun of Reactor Systems Branch. FICHE: 74378 287 acn: 9303250156
March 17, 1993	C.B. Brinkman, CE, letter forwarding summary of inservice testing requirements and listings of pumps and valves to be tested. FICHE: 74401 228 acn: 9303260154
March 23, 1993	C.B. Brinkman, CE, letter forwarding minor revisions to draft SER responses requested by NRC reviewer on Systems 80+ QA program, including revision of Table 3.2-1 to show graded quality classifications for Systems 80+ structures, systems and components. FICHE: 74422 193 acn: 9303290284
March 23, 1993	C.B. Brinkman, CE, letter forwarding design description and ITAAC for System 80+ nuclear island structure. FICHE: 74418 292 acn: 9303290287
March 23, 1993	C.B. Brinkman, CE, letter forwarding "System 80+ Severe Accident Phenomenology and Containment Performance" report. FICHE: 74421 001 acn: 9303300241
March 26, 1993	C.B. Brinkman, CE, letter forwarding revised safety analysis being submitted in response to draft issues, NRC staff questions and comments provided at December 1992 and January 1993 meetings. FICHE: 74468 001 acn: 9304020218
March 26, 1993	C.B. Brinkman, CE, letter forwarding Revision 1 to NPX80-IC-RR790-02, "Human Factors Evaluation and Allocation of System 80+ Functions." FICHE: 74456 056 acn: 9304020223

March 26, 1993	C.B. Brinkman, CE, letter forwarding Amendment M to Volumes 19 through 26 of "System 80+ Standard Design CESSAR Design Certification (DC)." FICHE: 74508 001 acn: 9304070207
March 29, 1993	C.B. Brinkman, CE, letter forwarding revisions to Chapters 2 and 3 of CESSAR-DC "System 80+ SAR - Seismic/Structural Changes." FICHE: 74459 001 acn: 9304050068
April 2, 1993	C.B. Brinkman, CE, letter forwarding results of levels 2 and 3 of System 80+ PRA and revised response to Open Item 19.1.2.1.1.8-1. FICHE: 74543 001 acn: 9304120090
April 2, 1993	C.B. Brinkman, CE, letter forwarding requested information regarding System 80+ fire protection and revised DSER responses. FICHE: 74568 001 acn: 9304120268
April 5, 1993	C.B. Brinkman, CE, letter forwarding listing of agreements and open issues compiled during four day review session and confirmed and modified in three hour wrap-up session with Russell. FICHE: 74601 001 acn: 9304130392
April 6, 1993	T.E. Murley, NRC, letter discusses review status of ABB-CE System 80+ design. Staff considers DSER for System 80+ equivalent to DFSER for General Electric Co. ABWR design. FICHE: 74557 358 acn: 9304120293
April 15, 1993	C.B. Brinkman, CE, letter confirms that full-size P&IDs of CESSAR-DC figures listed in inservice testing program, transmitted via March 17, 1993, letter to BNL to support NRC review of program. FICHE: 74746 351 acn: 9304260226
April 15, 1993	C.B. Brinkman, CE, letter forwarding markups to CESSAR-Design Certification for closure of corresponding DSER, including revisions to I&C system descriptions in Chapter 7 and revisions to fire protection system description in Section 9.5.1. FICHE: 74757 001 acn: 9304260289
April 15, 1993	C.B. Brinkman, CE, letter forwarding Amendment N to "CESSAR - Design Certification." FICHE: 74780 001 acn: 9304280111
April 16, 1993	R.W. Borchardt, NRC, discusses review of Human Factors Program Plan and operating experience review for System 80+ Design and request mfg address unresolved issues identified in plans within 60 days. FICHE: 74737 253 acn: 9304230053

April 20, 1993	C.B. Brinkman, CE, letter forwarding revised bases for System 80+ TS. FICHE: 74755 001 acn: 9304260290
April 21, 1993	C.B. Brinkman, CE, letter forwarding markups of CESSAR-DC for closure of DSER issues, including minor technical revisions to seven DSER issues being resolved by Plant System Branch as result of increase in core power level. FICHE: 74761 104 acn: 9304270356
April 29, 1993	I. Selin, NRC, letter responding to April 6, 1993, letter commenting on importance of completing reviews of both ABWR and System 80+ designs as quickly as possible. FICHE: 74835 084 acn: 9305060342
April 30, 1993	C.B. Brinkman, CE, letter forwarding supplemental information on System 80+ design descriptions, covering classification of structures, system and components, compartment pressurization and temperature analysis outside containment and station service water system structure. FICHE: 75884 135 acn: 9305110202
April 30, 1993	C.B. Brinkman, CE, letter forwarding Supplement 1 of System 80+ design descriptions and ITAAC (inspections, tests, analyses, and acceptance criteria). FICHE: 74880 271 acn: 9305110222
May 3, 1993	C.B. Brinkman, CE, letter forwarding additional System 80+ Submittal 1 design descriptions and ITAAC covering station service water system and component cooling water system. FICHE: 74884 312 acn: 9305110197
May 3, 1993	C.B. Brinkman, CE, letter forwarding information requested by NRC to supplement information on System 80+ design descriptions covering station service water and component cooling water system. FICHE: 74885 001 acn: 9305110221
May 10, 1993	A.M. Dibiasio, BNL, reviews CE System 80+ Inservice Testing Plan for FIN E-2024, Task 5. FICHE: 71706 340 acn: 9306030033
May 13, 1993	M.X. Franovich, NRC, provides summary of April 6, 1993, telcon conducted among representatives of NRC, BNL and ABB-CE. FICHE: 74997 230 acn: 9305240248
May 14, 1993	C.B. Brinkman, CE, letter forwarding Amendment O to "System 80+ CESSAR for Design Certification." FICHE: 74972 032 acn: 9305200369

May 14, 1993	C.B. Brinkman, CE, letter forwarding markups of CESSAR-DC for closure of Chapters 2 and 3 draft SER issues and structural design detail results. FICHE: 74989 001 acn: 9305240118
May 19, 1993	C.B. Brinkman, CE, letter forwarding "Common Mode Failure Evaluation for Limiting Fault Events," to closeout common mode failure issue for System 80+ RPS AND ESF instrumentation. FICHE: 75145 001 acn: 9306040125
May 26, 1993	J.R. Egan, CE, provides ABB-CE proposed System 80+ Tier 1 definition for "Channel." FICHE: 75613 359 acn: 9307080238
May 27, 1993	R.A. Matzie, CE, letter discussing review of SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary & ALWR Designs," dated April 2, 1993. FICHE: 75412 293 acn: 9306210122
May 28, 1993	C.B. Brinkman, CE, letter forwarding Submittal 2 of System 80+ design descriptions and associated ITAAC (inspections, tests, analyses, and acceptance criteria). FICHE: 75276 108 acn: 9306080218
May 28, 1993	C.B. Brinkman, CE, letter forwarding System 80+ supplement information on design descriptions. FICHE: 75276 001 acn: 9306080245
June 4, 1993	C.B. Brinkman, CE, letter forwarding information requested to supplement information on System 80+ design descriptions and ITAAC. FICHE: 75368 234 acn: 9306150180
June 4, 1993	C.B. Brinkman, CE, letter forwarding additional information on System 80+ design descriptions and associated ITAAC, per May 28, 1993. FICHE: 75366 334 acn: 9306150187
June 11, 1993	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses, including revised response to Open Item 5.2.3-3, markups to Section 6.2.5, 45 responses to Plant Systems Branch issues including fire protection and revised DSER responses. FICHE: 75475 001 acn: 9306250018
June 15, 1993	C.B. Brinkman, CE, letter forwarding "Evaluation of System 80+ Standard Design Interfacing System LOCA Challenges," special report improvements identified to System 80+ design as result of evaluation listed. FICHE: 75508 269 acn: 9306230181

June 15, 1993	C.B. Brinkman, CE, letter forwarding Amendment P to "System 80+ CESSAR - Design Certification." FICHE: 75480 001 acn: 9306240083
June 16, 1993	D. Crutchfield, NRC, letter responding to May 27, 1993, letter providing preliminary comments on SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and ALWR Designs." FICHE: 75412 289 acn: 9306210116
June 16, 1993	D. Crutchfield, NRC, letter forwarding two technical reports requested by CE technical staff during recent meetings with NRC technical staff members. FICHE: 75412 289 acn: 9306210116
June 18, 1993	C.B. Brinkman, CE, letter forwarding Revision 1 to "Design Alternatives to System 80+ Nuclear Power Plant," covering cost/benefit based on recently revised PRA. FICHE: 75509 077 acn: 9306280248
June 18, 1993	C.B. Brinkman, CE, letter forwarding response to NRC request for additional information to supplement information on System 80+ design descriptions. FICHE: 75569 279 acn: 9307020299
June 18, 1993	C.B. Brinkman, CE, letter forwarding submittal 3 of System 80+ design descriptions and associated ITAAC for review and approval. FICHE: 75584 180 acn: 9307020344
June 24, 1993	C.B. Brinkman, CE, letter requests summary assessment of NRC present position on licensibility status of Nuplex 80+ design. FICHE: 75714 318 acn: 9307200339
June 25, 1993	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses. FICHE: 75595 241 acn: 9307060210
June 29, 1993	C.B. Brinkman, CE, letter forwarding supplemental information on System 80+ design description. FICHE: 75625 334 acn: 9307090277
June 29, 1993	C.B. Brinkman, CE, letter forwarding submittal 4 of System 80+ design descriptions and associated ITAAC, in conjunction with June 29, 1993 letter. FICHE: 75627 186 acn: 9307090280
June 29, 1993	C.B. Brinkman, CE, letter forwarding set of design descriptions and inspections tests, analyses, and acceptance criteria for ITAAC System 80+ for initial test program. FICHE: 75712 326 acn: 9307160174

July 14, 1993	C.B. Brinkman, CE, letter forwarding proprietary System 80+ drawings on structures and equipment arrangements. FICHE: 75952 241 acn: 9307220206
July 15, 1993	S.L. Magruder, NRC, letter forwarding two technical reports requested by CE technical staff during recent meetings with NRC technical staff members. FICHE: 75979 001 acn: 9308040086
July 16, 1993	C.B. Brinkman, CE, letter forwarding Amendment Q to "CESSAR - Design Certification (DC)," incorporating revisions previously transmitted as draft mark-up pages of CESSAR-DC and other material discussed with NRC at recent meetings. FICHE: 75775 001 acn: 9307210285
July 16, 1993	C.B. Brinkman, CE, letter forwarding responses to five draft SER follow-on questions on PRA, information on piping analysis spectra, additional information to close items from June 21 through 25 piping audit and information for waterhammer benchmark analysis for closure of questions to draft SER. FICHE: 76310 154 acn: 9309010139
July 22, 1993	C.B. Brinkman, CE, letter forwarding Revision 1 to Volume 1 of "System 80+ Design Certification Fire Hazards Assessment (FHA)," per March 26, 1993, submittal and April 26 through 30, 1993, meeting. FICHE: 75806 262 acn: 9307270091
July 23, 1993	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses. FICHE: 75988 001 acn: 9308030004
July 23, 1993	C.B. Brinkman, CE, letter forwarding response to NRC May 17 and June 8, 1993, RAIs on CESSAR Design Certification. FICHE: 75991 096 acn: 9308040154
July 23, 1993	D. Crutchfield, NRC, letter responding to June 24, 1993, request for status of NRC review and general assessment of Nuplex 80+ advanced control complex design for System 80+. FICHE: 76064 344 acn: 9308110337
July 26, 1993	C.B. Brinkman, CE, letter forwarding SG Parameters requested by A. Thadani at July 15, 1993, senior management meeting in Windsor, Connecticut. FICHE: 76570 295 acn: 9309270027
July 29, 1993	C.B. Brinkman, CE, letter forwarding information for closure of System 80+ draft SER open issues, in response to June 14 and 15, and July 7 and 8, 1993, meetings with NRC and response to RAI 440.230. FICHE: 76073 001 acn: 9308100099

July 30, 1993	T.V. Wambach, NRC, letter forwarding preliminary comments on technical specifications System 80+ (Chapter 16 CESSAR-DC) submitted in Appendix K. FICHE: 76151 251 acn: 9308190043
August 11, 1993	C.K. Tang, letter forwarding ABWR STS 3.1.7, "Standby Liquid Control Systems." FICHE: 76154 312 acn: 9308180046
August 12, 1993	D. Crutchfield, NRC, letter requesting that ABB-CE comprehensively and systematically evaluate potential design alternatives proposed to reduce probability of SGTR containment bypass accident sequences. Fiche: 76220 239 acn: 9308270065
August 13, 1993	C.B. Brinkman, CE, letter submitting assessment of potential for changes to Tier 1 design descriptions and inspections, tests, analysis and acceptance criteria (DD/ITAAC), submitted from April 30 through June 30, 1993. FICHE: 76206 319 acn: 9308230173
August 25, 1993	C.B. Brinkman, CE, letter forwarding CESSAR-design certification Amendment Q overview, per July 16, 1993, submittal. FICHE: 76341 154 acn: 9309020217
August 26, 1993	D. Crutchfield, NRC, letter responding to informal inquiries from design certification applicants regarding form and content of design control document. FICHE: 76396 347 acn: 9309100241
August 31, 1993	C.B. Brinkman, CE, letter forwarding overview of Amendment R to "Standard SAR - Design Certification" and affidavit per 10 CFR 50.4(b) and 50.30(b). FICHE: 76394 342 acn: 9309080190
August 31, 1993	C.B. Brinkman, CE, letter forwarding formally printed Amendment R to "Standard SAR - Design Certification (CESSAR-DC)," with two oversized figures. FICHE: 76375 001 acn: 9309080313
September 1, 1993	C.B. Brinkman, CE, letter forwarding Appendix 3.7C, describing soil structure interaction analysis, information for Reactor System Branch and editorial changes regarding Systems 80+ information for issue closure. FICHE: 76453 001 acn: 9309150359
September 2, 1993	R.W. Borchardt, NRC, forwards NRC comments on CE System 80+ Tier 1 submittals dated April 30, May 3 through 8, June 4, June 18, and June 29, 1993. FICHE: 76461 024 acn: 9309160235

September 23, 1993	C.B. Brinkman, CE, letter forwarding Attachments 2 through 4 and 9 through 14, providing material to close follow-on questions to DSER responses regarding System 80+ TM information for issue closure. FICHE: 76627 001 acn: 9310010036
September 24, 1993	C.B. Brinkman, CE, letter forwarding information on Human Factors Engineering verification and validation, including CESSAR-DC markups for main control room procedures validation. FICHE: 76633 129 acn: 9310010219
September 28, 1993	W.H. Rasin, NUMARC, letter forwarding revision 2 to NPX80-IC-DP790-01, "Human Factors Program Plan for System 80+ (TM) Std Plant Design," "Plant Designers Operational Support Information Plan. . ." and Revision 2 to "Design Alternatives for System 80+ Nuclear Power Plant. FICHE: 71886;002-71886:026 acn: 93100703564
September 28, 1993	W.H. Rasin, NUMARC, letter providing comments on draft guidance on form and content of design control document. FICHE: 71886 002 acn: 9310070374
September 28, 1993	T.V. Wambach, NRC, requests response to enclosure comments to continue review of technical specifications for System 80+. Informs that deviation from STS (NUREG-1432). FICHE: 76705 221 acn: 9310070097
September 28, 1993	M.X. Franovich, NRC, forwards preliminary draft version of NUREG-CR-6105, "Human Factors Engineering Guidelines for Review of Advanced Alarm Systems" and draft of NUREG-CR-5908, "Advanced Human System Interface Design Review Guideline." FICHE: 76708 001 acn: 9310070253
September 30, 1993	C.B. Brinkman, CE, letter forwarding Revision 2 to NPX80-IC-DP790-01, "Human Factors Program Plan for System 80+ (TM) Standard Plant Design," "Plant Designers Operational Support Information Plan. . .," and Revision 2 to "Design Alternatives for System 80+ Nuclear Power Plant." FICHE: 76718 001 acn: 9310070357
October 1, 1993	T.V. Wambach, NRC, letter forwarding comments regarding ABB-CE software program manual for System 80+. FICHE: 71886 203 acn: 9310070196
October 6, 1993	C.B. Brinkman, CE, letter forwarding "Evaluation of System 80+ Standard Design for SG Tube Rupture Events." FICHE: 76781 001 acn: 9310140185

October 11, 1993	C.B. Brinkman, CE, letter forwarding TS markups identifying differences form STS (NUREG-1432) and basis for differences. FICHE: 77017 001 acn: 9310280008
October 14, 1993	T.V. Wambach, NRC, letter forwarding "LLNL Comments on ABB-CE System 80+ I&C Technical Specifications." FICHE: 76821 239 acn: 9310190260
October 18, 1993	C.B. Brinkman, CE, letter forwarding human factors engineering issues regarding response to cross-branch Chapter 18 question and design process requirements A-3.6 availability verification. FICHE: 77107 340 acn: 9311040383
October 20, 1993	C.B. Brinkman, CE, letter forwarding markups of System 80+ Design Descriptions, associated ITAAC and CESSAR-DC pages form meetings with NRC staff on October 4, 1993. FICHE: 76920 072 acn: 9310260242
October 27, 1993	C.B. Brinkman, CE, letter forwarding draft Appendix 19.11K, "Assessment of System 80+ Hydrogen Mitigation System for Application in Severe Accident Environmental" and other material to close follow-on questions to draft SER responses. FICHE: 77104 002 acn: 9310040313
October 29, 1993	C.B. Brinkman, CE, letter forwarding Amendment S to CESSAR - Design Certification, per 10 CFR 50.4(b) and 10 CFR 50.30(b). FICHE: 77063 001 acn: 9311040120
November 3, 1993	C.B. Brinkman, CE, letter forwarding material closing follow-up questions to DSER responses regarding System 80+ information. FICHE: 77160 001 acn: 9311100097
November 4, 1993	C.B. Brinkman, CE, letter forwarding Systems 80+ information for closure of follow-on questions to draft SER responses. FICHE: 77216 001 acn: 9311150079
November 4, 1993	D. Crutchfield, NRC, provides update on status of review of Nuplex 80+ control complex for System 80+ design certification. FICHE: 77168 001 acn: 9311120014
November 5, 1993	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses and CESSAR-DC revisions from ITAAC revisions to IST program, information on seismic I tanks and remaining four emergency guidelines for standard recovery actions. FICHE: 77184 001 acn: 9311150099

November 12, 1993	C.B. Brinkman, CE, letter forwarding marked-up changes to System 80+ DD/ITAAC and ITAAC-related change pages reflecting forthcoming CESSAR-DC Amendment T. FICHE: 77303 001 acn: 9312020137
November 12, 1993	C.B. Brinkman, CE, letter forwarding Revision 1 to NPX80-SQP-0101.0, "Software Program Manual for Nuplex 80+" and draft revision to System 80+ Emergency Operations Guidelines, "Functional Recovery Guideline" to close follow-up question to DSER responses. FICHE: 77321 143 acn: 9312020409
November 15, 1993	C.B. Brinkman, CE, letter forwarding corrected pages for Chapter 19. FICHE: 77314 321 acn: 9312020229
November 16, 1993	T.V. Wambach, NRC, letter forwarding comments on technical specifications for System 80+; requests responses no later than two weeks from letter receipt to maintain review schedules. FICHE: 77287 187 acn: 9311300097
November 19, 1993	C.B. Brinkman, CE, letter forwarding text portion for CESSAR-DC Section 143, addressing basis for selecting Tier 1 design certification material and associate ITAAC for comment. FICHE: 77310 304 acn: 9312020185
November 23, 1993	S.M. Long, NRC, letter forwarding comments that address operator actions during SGTR sequences, both with success and failure of SI function for review. FICHE: 71963 019 acn: 9312080118
November 24, 1993	B.A. Boger, NRC, commends outstanding work of J. O'Hara on System 80+ advanced reactor human factors review project. FICHE: 71943 203 acn: 9312090127
November 29, 1993	C.B. Brinkman, CE, letter forwarding Amendment T to "CE Standard SAR - Design Certification" and affidavit required by 10 CFR 50.4(b) and 30(b). FICHE: 77389 001 acn: 9312070257
December 3, 1993	T.H. Boyce, NRC, letter forwarding initial staff comments on ABB-CE System 80+ Tier 1 October 20, 1993, and November 12, 1993, submittals. FICHE: 77490 244 acn: 9312160233
December 7, 1993	T.V. Wambach, NRC, letter forwarding staff comments on markup copy of affected technical specifications and requests that responses to comments be provided no later than two weeks from receipt of letter. FICHE: 77493 001 acn: 9312160300

December 14, 1993	C.B. Brinkman, CE, letter responding to request of NRC staff and forwards five copies of selected CESSAR-DC parameters and assumptions addressed in certified design material for System 80+ standard plant design. FICHE: 77565 186 acn: 9312200315
December 17, 1993	C.B. Brinkman, CE, letter providing material to close follow-on to DSER responses regarding System 80+ information for issue closure. FICHE: 77677 274 acn: 9312300221
December 31, 1993	C.B. Brinkman, CE, responds to comments on ABB-CE design descriptions and associated inspections, tests, analyses and acceptance criteria and on changes to CESSAR-DC. FICHE: acn: 9401050317
January 7, 1994	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses, including list of ALWR Utility Requirements Documents deviations and three tables of cross references of CESSAR-DC. FICHE: 77912 020 acn: 9401190026
January 7, 1994	K.M. Shembarger, NRC, letter forwarding comments on TS Sections 4 and 5 for System 80+ markup copy of affected TS enclosure. FICHE: 77797 337 acn: 9401140007
January 10, 1994	K.M. Shembarger, NRC, letter forwarding summary of December 21, 1993, meeting with utility in Rockville, Maryland regarding new source term and application for EQ of System 80+ plant. FICHE: 77833 268 acn: 9401130208
January 10, 1994	S.L. Magruder, NRC, letter forwarding summary of December 14, 1993, meeting with licensees in Rockville, Maryland to review status of all outstanding issues in civil and geosciences area and discuss schedule for closure of remaining issues. FICHE: 77824 264 acn: 9401140201
January 12, 1994	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses regarding System 80+ information. FICHE: 77929 001 acn: 9401250287
February 1, 1994	T.H. Boyce, NRC, letter forwarding summary of meeting with DOE in Rockville, Maryland to discuss progress of reviews for design certification of next-generation-reactor-designs. FICHE: 78106292 acn: 9402140161
February 8, 1994	C.B. Brinkman, CE, letter forwarding Amendment U to "CESSAR-DC." FICHE: 78241 001 acn: 9402180240

February 8, 1994	C.B. Brinkman, CE, letter forwarding "System 80+ Emergency Operations Guidelines." FICHE: 78225 001 acn: 9402220103
February 16, 1994	K.M. Shembarger, NRC, letter discussing NRC staff review of System 80+ TS, including applicable portions of Amendment U. FICHE: 78217 001 acn: 9402240158
February 21, 1994	C.B. Brinkman, CE, letter forwarding revision to CESSAR-DC documenting resolution of NRC concern on small break LOCA with boron dilution. FICHE: 78438 272 acn: 9403100249
February 22, 1994	C.B. Brinkman, CE, letter forwarding listed material to close follow-on questions to DSER responses regarding System 80+. FICHE: 78419 001 acn: 9403080173
February 24, 1994	C.B. Brinkman, CE, letter forwarding revision to CESSAR-DC and material to close follow-on questions to DSER responses regarding System 80+. FICHE: 78433 001 acn: 9403080284
February 25, 1994	C.B. Brinkman, CE, letter forwarding material to close follow-on questions to DSER responses regarding System 80+. FICHE: 78432 276 acn: 9403080264
February 28, 1994	T.E. Murley, NRC, letter informing of approval of FSER on ABB-CE System 80+ design. FICHE: 78364 026 acn: 9403070250
March 1, 1994	K.M. Shembarger, NRC, letter submitting comments on System 80+ TS. FICHE: 78410 001 acn: 9403100042
March 3, 1994	R.W. Borchardt, NRC, letter submitting comments on System 80+. FICHE: 78410 001 acn: 9403100042
March 3, 1994	R.W. Borchardt, NRC, letter forwarding advance copy of FSER regarding NRC review of application for certification of System 80+ design. FICHE: 78462 001 acn: 9403140197
March 4, 1994	C.B. Brinkman, CE, letter supplements December 31, 1993, letter submitted by EPRI, forwarding report on technical aspects of emergency planning for ALWRs. FICHE: 78587 327 acn: 9403180262
March 8, 1994	R.W. Borchardt, NRC, letter forwarding advance copy of FSER regarding NRC revciew of application for certification of System 80+ design. FICHE: 78462 001 acn: 9403140197

March 24, 1994	R.W. Borchardt, NRC, letter requesting comments on enclosure CDM review guidance by May 20, 1994. FICHE: 78686 306 acn: 9403290295
March 24, 1994	T.H. Boyce, NRC, letter providing comments on CDM and CESSAR-DC, inspections, tests, analyses and acceptance criteria ITAAC task group. FICHE: 78727 164 acn: 9404010018
March 25, 1994	C.B. Brinkman, CE, letter forwarding SG thermal-hydraulic summary analysis results. FICHE: 78913 085 acn: 9404150115
March 30, 1994	D.A. Dreyfus, DOE, letter forwarding draft "Advanced Reactor Research and Development Program 5-yr Plan for Advanced Reactor Activities Under Energy Policy Act of 1992." FICHE: 940330 acn: 9404250187
April 12, 1994	C.B. Brinkman, CE, letter informing NRC of change in organizational alignment of ABB-CE Nuclear Fuels, which has become part of ABB-CE Nuclear Operations. FICHE: 78873 285 acn: 9404150063
April 13, 1994	R.W. Borchardt, NRC, letter forwarding current copy of Chapter 20, "Generic Issues," of FSER for System 80+ design. FICHE: 79020 001 acn: 9404270258
April 14, 1994	D.A. Dreyfus, DOE, letter informing that deadline for stakeholder comments on "draft 5-year Plan for Advanced Reactor Activities Under Energy Policy Act of 1992," extended until May 2, 1994. FICHE: 79004 345 acn: 9404250174
April 14, 1994	C.B. Brinkman, CE, letter forwarding revision to CESSAR-DC and comments on System 80+ FSER, providing additional structural design detail and technical revision to TS agreed upon at April 5 and 6, 1994, meeting. FICHE: 79063 001 acn: 9404290296
April 26, 1994	J.M. Taylor, NRC, letter submitting comments regarding draft report "5-year Plan for Advanced Reactor Activities Under Energy Policy Act of 1992." FICHE: 79078 237 acn: 9405030173
April 26, 1994	C.B. Brinkman, CE, letter forwarding summary of System 80+ design and operational features involving deviations from current regulations. FICHE: 79243 316 acn: 9405060298
April 26, 1994	C.B. Brinkman, CE, letter providing material to close issues raised by NRC staff, including summary of approach to design verification and list of calculations. FICHE: 79214 289 acn: 9405060301

April 29, 1994	C.B. Brinkman, CE, letter forwarding copies of Amendment V to "CESSAR-DC." FICHE: 79090 001 acn: 9405020194
April 29, 1994	C.B. Brinkman, CE, letter forwarding response to NRC comments on ABB-CE design certification material transmitted with March 24, 1994, letter. FICHE: 79298 001 acn: 9405110237
April 29, 1994	C.B. Brinkman, CE, letter forwarding revisions to CESSAR-DC and comments on System 80+ FSER. FICHE: 79777 001 acn: 9406140384
May 2, 1994	C.B. Brinkman, CE, letter advising that enclosed report KVB 75-642, "Experimental Investigation of Relief Valve Vent Clearing Phenomena," supporting System 80+ IRWST design activities should be withheld. FICHE: 79312 356 acn: 9405100077
May 10, 1994	C.B. Brinkman, CE, letter forwarding "Technical Support Document for Amendments to 10 CFR Part 51 Considering Severe Accident Under NEPA for Plants of System 80+ design." FICHE: 79387 167 acn: 9405170173
May 11, 1994	C.B. Brinkman, CE, letter forwarding comments of Chapter 4 of System 80+ final SER, NUREG-1462 for technical accuracy and consistency with CESSAR-DC. FICHE: 79456 129 acn: 9405230166
May 15, 1994	A. Behababani, NRC, letter requestings confirmation on whether Scientech can execute 2-D Pm-ALPHA/ESPROSE codes including 2-D FCI calculations on System 80+ in letter reporting by June 15, 1994. FICHE: 72165 360 acn: 9406200192
May 18, 1994	C.B. Brinkman, CE, amends April 12, 1994 letter, informing NRC of change in organizational alignment of ABB-CE Nuclear Fuels, which has become part of outlining new organization encl. FICHE: 79622 295 acn: 9406020294
May 19, 1994	C.B. Brinkman, CE, letter informing that vendor providing detailed markup of draft document entitled, "Certified Design Material and ITAAC Review Guidance," to NEI, in response to NRC March 24, 1994, letter. FICHE: 79622 297 acn: 9406020293
May 24, 1994	R.W. Borchardt, NRC, letter forwarding inspection report 99900401/94-01 on February 14 through 18, 1994 and notice of nonconformance. FICHE: 79709 152 acn: 9406080255

June 1, 1994	C.B. Brinkman, CE, letter responding to question asked by NRC regarding controls listed in System 80+ certified design material Table 2.12.1-1, "MCR Min Inventory of Fixed Position Annunciators, Displays and Controls." FICHE: 79679 163 acn: 9406070176
June 10, 1994	C.B. Brinkman, CE, letter responding to question asked by NRC regarding controls listed in System 80+ certified design material Table 2.12.1-1, "MCR Min Inventory of Fixed Position Annunciators, Displays and Controls." FICHE: 79679 163 acn: 9406070176
June 10, 1994	C.B. Brinkman, CE, letter forwarding revisions to selected CESSAR-DC parameters and assumptions addressed in certified design material, provided by ABB-CE December 14, 1993, letter. FICHE: 80019 326 acn: 9406280284
June 16, 1994	C.B. Brinkman, CE, letter responding to May 25, 1994, letter regarding violations noted in nonconformance Inspection Report 99900401/94-01. FICHE: 79878 186 acn: 9406220335
June 17, 1994	C.B. Brinkman, CE, letter forwarding Revision O to "System 80+ Emergency Operations Guidelines." FICHE: 79899 001 acn: 9406220370
June 20, 1994	C.B. Brinkman, CE, letter forwarding revised Combustion Engineering Nuclear Fuel. FICHE: 80016 315 acn: 9406290022
June 20, 1994	C.B. Brinkman, CE, letter forwarding revised System 80+ certified design material and affidavit. FICHE: 79981 001 acn: 9406220379
July 6, 1994	K.M. Shembarger, NRC, letter forwarding documentation of conference call between NRC and ABB-CE on July 23, 1994, to resolve staff comments on Amendment W to ABB-CE System 80+ TS. FICHE: 80207 357 acn: 9407130265
July 8, 1994	D.M. Crutchfield, NRC, letter advising that design reliability assurance program Tier 1 information and ITAAC developed in 1993 be used as starting point for development of ITAAC to be included in certified design material. FICHE: 80188 335 acn: 9407120161
July 12, 1994	C.B. Brinkman, CE, letter forwarding certified design material related to design reliability assurance program and corresponding errata pages for CESSAR-DC (Section 1.0, 17.3 and 19.15) Amendment W, to resolve NRC comments received by telcon over past two weeks. FICHE: 80315 194 acn: 9407210061

July 18, 1944	R.W. Borchardt, NRC, letter approving licensee request for withholding of System 80+ drawings and electronic diskette with design and operations data for MAAP computer code from public disclosure. FICHE: 80307 333 acn: 9407210348
July 20, 1994	C.B. Brinkman, CE, letter requesting that service list for Docket No. 52-002 be updated in accordance with enclosed listing, due to changes in position titles, affiliations and organizations. FICHE: 80335 333 acn: 9407250147
July 20, 1994	C.B. Brinkman, CE, letter seeking confirmation of vendor understanding of NRC fee regulations in 20 CFR Part 170 as pertaining to System 80+ design certification rulemaking as set forth in enclosure. FICHE: 80339 344 acn: 9407250204
July 20, 1994	R.W. Borchardt, NRC, letter informing that proprietary reports NPX80-IC-SD790-02, RI and KVB 75-643 removed from docket and destroyed. FICHE: 80328 353 acn: 9407250314
July 21, 1994	C.B. Brinkman, CE, letter forwarding errata to System 80+ certified design material, in response to editorial and word processing errors. FICHE: 80415 243 acn: 9407290346
July 26, 1994	W.T. Russell, NRC, letter transmitting FDA for standard System 80+ design, per Appendix O of 10 CFR Part 52. FICHE: 80366 001 acn: 9407280072
July 26, 1994	R.W. Borchardt, NRC, letter responding to letter dated March 25, 1994, requesting that thermal-hydraulic summary analysis results on nuclear plant SGs (System 80+) be deemed proprietary and withheld from public disclosure. FICHE: 80423 310 acn: 940820147

APPENDIX B

INDEX OF NRC REQUESTS FOR ADDITIONAL INFORMATION AND CE RESPONSES

RAI NUMBER	DATE NRC LETTER	RESPONSE	DATE
1	12/08/87	LD-88-021	3/22/88
1	12/08/87	LD-88-091	9/14/88
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250.3	03/11/88	LD-88-039	6/6/88
410.1	03/15/88	LD-88-046	6/30/88
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410.5	03/15/88	LD-88-046	6/30/88
410.6	03/15/88	LD-88-046	6/30/88
410.7	03/15/88	LD-88-046	6/30/88
410.8	03/15/88	LD-88-046	6/30/88
410.9	03/15/88	LD-88-046	6/30/88
410.10	03/15/88	LD-88-046	6/30/88
410.11	03/15/88	LD-88-046	6/30/88
420.1	04/13/88	LD-88-052	7/13/88
420.2	04/13/88	LD-88-052	7/13/88
420.3	04/13/88	LD-88-052	7/13/88
500.4	06/01/88	LD-88-068	8/1/88
500.5	06/01/88	LD-88-068	8/1/88
500.6	06/01/88	LD-88-068	8/1/88
218-14	06/28/88	LD-88-099	9/20/88
218-15	06/28/88	LD-88-099	9/20/88
218-16	06/28/88	LD-88-099	9/20/88
218-17	06/28/88	LD-88-099	9/20/88
218-18	06/28/88	LD-88-099	9/20/88
281-19	06/28/88	LD-88-099	9/20/88
281-20	06/28/88	LD-88-099	9/20/88
281-21	06/28/88	LD-88-099	9/20/88
281-22	06/28/88	LD-88-099	9/20/88
281-23	06/28/88	LD-88-099	9/20/88
281-24	06/28/88	LD-88-099	9/20/88
440-1	06/28/88	LD-88-089	9/9/88
440-2	06/28/88	LD-88-089	9/9/88
281.50	08/03/88	LD-88-151	12/7/88
281.51	08/03/88	LD-88-151	12/7/88
281.52	08/03/88	LD-88-151	12/7/88
281.53	08/03/88	LD-88-151	12/7/88
281.54	08/03/88	LD-88-151	12/7/88
281.55	08/03/88	LD-88-151	12/7/88
281.56	08/03/88	LD-88-151	12/7/88
410.12	08/03/88	LD-88-151	12/7/88
410.13	08/03/88	LD-88-151	12/7/88
410.14	08/03/88	LD-88-151	12/7/88
410.15	08/03/88	LD-88-151	12/7/88
410.16	08/03/88	LD-88-151	12/7/88
260.1.A	08/02/88	LD-88-119	10/21/88
260.2.A	08/02/88	LD-88-119	10/21/88
260.6.A	08/02/88	LD-88-119	10/21/88
260.10.A	08/02/88	LD-88-119	10/21/88
	10/11/88	LD-89-029	3/17/89
410.17	10/20/88	LD-89-091	8/16/89
410.18	10/20/88	LD-89-091	8/16/89
410.19	10/20/88	LD-89-091	8/16/89
410.20	10/20/88	LD-89-091	8/16/89

410.21	10/20/88	LD-89-091	8/16/89
410.22	10/20/88	LD-89-091	8/16/89
410.23	10/20/88	LD-89-091	8/16/89
410.24	10/20/88	LD-89-091	8/16/89
410.25	10/20/88	LD-89-091	8/16/89
410.26	10/20/88	LD-89-091	8/16/89
410.27	10/20/88	LD-89-091	8/16/89
410.28	10/20/88	LD-89-091	8/16/89
410.29	10/20/88	LD-89-091	8/16/89
410.30	10/20/88	LD-89-091	8/16/89
410.31	10/20/88	LD-89-091	8/16/89
251.11	10/26/88	LD-89-110	10/4/89
251.11	10/26/88	LD-89-110	10/4/89
251.11	10/26/88	LD-89-110	10/4/89
251.11	10/26/88	LD-89-110	10/4/89
251.11	10/26/88	LD-89-110	10/4/89
251.11	10/26/88	LD-89-110	10/4/89
251.0	10/28/88	LD-89-106	9/19/89
251.1	10/28/88	LD-89-106	9/19/89
250.2	10/28/88	LD-89-106	9/19/89
250.3	10/28/88	LD-89-106	9/19/89
251.4	10/28/88	LD-89-106	9/19/89
251.5	10/28/88	LD-89-106	9/19/89
251.6	10/28/88	LD-89-106	9/19/89
251.7	10/28/88	LD-89-106	9/19/89
251.8	10/28/88	LD-89-106	9/19/89
251.9	10/28/88	LD-89-106	9/19/89
251.10	10/28/88	LD-89-106	9/19/89
281-25	11/01/88	LD-89-079	7/21/89
281-26	11/01/88	LD-89-079	7/21/89
281-27	11/01/88	LD-89-079	7/21/89
281-28	11/01/88	LD-89-079	7/21/89
281-29	11/01/88	LD-89-079	7/21/89
500.7	11/02/88	LD-89-107	9/28/89
500.8	11/02/88	LD-89-107	9/28/89
500.9	11/02/88	LD-89-107	9/28/89
500.10	11/02/88	LD-89-107	9/28/89
500.11	11/02/88	LD-89-107	9/28/89
500.12	11/02/88	LD-89-107	9/28/89
3.1	12/15/88	LD-89-070	7/6/89
3.2	12/15/88	LD-89-070	7/6/89
3.3	12/15/88	LD-89-070	7/6/89
1	12/15/88	LD-89-070	7/6/89
2	12/15/88	LD-89-070	7/6/89
3	12/15/88	LD-89-070	7/6/89
4	12/15/88	LD-89-070	7/6/89
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90

252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
252.1	12/16/88	LD-90-088	1/25/90
1	01/23/89	LD-89-117	10/30/89
2	01/23/89	LD-89-117	10/30/89
3	01/23/89	LD-89-117	10/30/89
4	01/23/89	LD-89-117	10/30/89
5	01/23/89	LD-89-117	10/30/89
6	01/23/89	LD-89-117	10/30/89
7	01/23/89	LD-89-117	10/30/89
8	01/23/89	LD-89-117	10/30/89
9	01/23/89	LD-89-117	10/30/89
10	01/23/89	LD-89-117	10/30/89
11	01/23/89	LD-89-117	10/30/89
12	01/23/89	LD-89-117	10/30/89
13	01/23/89	LD-89-117	10/30/89
14	01/23/89	LD-89-117	10/30/89
15	01/23/89	LD-89-117	10/30/89
16	01/23/89	LD-89-117	10/30/89
17	01/23/89	LD-89-117	10/30/89
18	01/23/89	LD-89-117	10/30/89
19	01/23/89	LD-89-117	10/30/89
20	01/23/89	LD-89-117	10/30/89
21	01/23/89	LD-89-117	10/30/89
22	01/23/89	LD-89-117	10/30/89
23	01/23/89	LD-89-117	10/30/89
24	01/23/89	LD-89-117	10/30/89
25	01/23/89	LD-89-117	10/30/89
26	01/23/89	LD-89-117	10/30/89
27	01/23/89	LD-89-117	10/30/89
28	01/23/89	LD-89-117	10/30/89
29	01/23/89	LD-89-117	10/30/89
30	01/23/89	LD-89-117	10/30/89
31	01/23/89	LD-89-117	10/30/89
32	01/23/89	LD-89-117	10/30/89
33	01/23/89	LD-89-117	10/30/89
34	01/23/89	LD-89-117	10/30/89
1	03/14/89	LD-89-092	8/17/89
2	03/14/89	LD-89-092	8/17/89
3	03/14/89	LD-89-092	8/17/89
4	03/14/89	LD-89-092	8/17/89
281-32	06/26/89	LD-91-013	3/15/91
281-33	06/26/89	LD-91-013	3/15/91
281-34	06/26/89	LD-91-013	3/15/91
281.57	06/26/89	LD-91-013	3/15/91
281.58	06/26/89	LD-91-013	3/15/91
281.59	06/26/89	LD-91-013	3/15/91

281.60	06/26/89	LD-91-013	3/15/91
281.61	06/26/89	LD-91-013	3/15/91
410.47	06/26/89	LD-91-013	3/15/91
410.48	06/26/89	LD-91-013	3/15/91
410.49	06/26/89	LD-91-013	3/15/91
410.50	06/26/89	LD-91-013	3/15/91
410.51	06/26/89	LD-91-013	3/15/91
410.52	06/26/89	LD-91-013	3/15/91
410.53	06/26/89	LD-91-013	3/15/91
410.54	06/26/89	LD-91-013	3/15/91
410.55	06/26/89	LD-91-013	3/15/91
410.56	06/26/89	LD-91-013	3/15/91
410.57	06/26/89	LD-91-013	3/15/91
410.58	06/26/89	LD-91-013	3/15/91
410.59	06/26/89	LD-91-013	3/15/91
410.60	06/26/89	LD-91-013	3/15/91
410.61	06/26/89	LD-91-013	3/15/91
410.62	06/26/89	LD-91-013	3/15/91
440.5	06/26/89	LD-91-013	3/15/91
440.6	06/26/89	LD-91-013	3/15/91
440.7	06/26/89	LD-91-013	3/15/91
440.8	06/26/89	LD-91-013	3/15/91
440.9	06/26/89	LD-91-013	3/15/91
471.1	06/26/89	LD-91-013	3/15/91
471.2	06/26/89	LD-91-013	3/15/91
471.3	06/26/89	LD-91-013	3/15/91
471.4	06/26/89	LD-91-013	3/15/91
471.8	06/26/89	LD-91-013	3/15/91
471.9	06/26/89	LD-91-013	3/15/91
471.11	06/26/89	LD-91-013	3/15/91
471.12	06/26/89	LD-91-013	3/15/91
471.13	06/26/89	LD-91-013	3/15/91
471.14	06/26/89	LD-91-013	3/15/91
471.15	06/26/89	LD-91-013	3/15/91
480.7	06/26/89	LD-91-018	4/29/91
500.13	06/26/89	LD-91-013	3/15/91
500.14	06/26/89	LD-91-013	3/15/91
810.1	06/26/89	LD-91-013	3/15/91
440.3	12/23/88	LD-91-010	3/4/91
440.4	12/23/88	LD-91-010	3/4/91
281-30	12/23/88	LD-91-010	3/4/91
281-31	12/23/88	LD-91-010	3/4/91
240.1	01/19/89	LD-91-012	3/15/91
270.1	01/19/89	LD-91-012	3/15/91
410.32	01/19/89	LD-91-012	3/15/91
410.33	01/19/89	LD-91-012	3/15/91
410.34	01/19/89	LD-91-012	3/15/91
410.35	01/19/89	LD-91-012	3/15/91
410.36	01/19/89	LD-91-012	3/15/91
410.37	01/19/89	LD-91-012	3/15/91
410.38	01/19/89	LD-91-012	3/15/91
410.39	01/19/89	LD-91-012	3/15/91
410.40	01/19/89	LD-91-012	3/15/91
410.41	01/19/89	LD-91-012	3/15/91
410.42	01/19/89	LD-91-012	3/15/91

410.43	01/19/89	LD-91-012	3/15/91
410.44	01/19/89	LD-91-012	3/15/91
410.45	01/19/89	LD-91-012	3/15/91
410.46	01/19/89	LD-91-012	3/15/91
480.1	01/19/89	LD-91-012	3/15/91
480.2	01/19/89	LD-91-012	3/15/91
480.3	01/19/89	LD-91-012	3/15/91
480.4	01/19/89	LD-91-012	3/15/91
480.5	01/19/89	LD-91-012	3/15/91
480.6	01/19/89	LD-91-012	3/15/91
280.1	01/24/90	LD-91-014	3/26/91
410.63	01/24/90	LD-91-014	3/26/91
410.64	01/24/90	LD-91-014	3/26/91
410.65	01/24/90	LD-91-014	3/26/91
410.66	01/24/90	LD-91-014	3/26/91
410.67	01/24/90	LD-91-014	3/26/91
410.68	01/24/90	LD-91-014	3/26/91
410.69	01/24/90	LD-91-014	3/26/91
410.70	01/24/90	LD-91-014	3/26/91
410.71	01/24/90	LD-91-014	3/26/91
410.72	01/24/90	LD-91-014	3/26/91
410.73	01/24/90	LD-91-014	3/26/91
410.74	01/24/90	LD-91-014	3/26/91
410.75	01/24/90	LD-91-014	3/26/91
410.76	01/24/90	LD-91-014	3/26/91
410.77	01/24/90	LD-91-014	3/26/91
410.78	01/24/90	LD-91-014	3/26/91
410.79	01/24/90	LD-91-014	3/26/91
410.80	01/24/90	LD-91-014	3/26/91
410.81	01/24/90	LD-91-014	3/26/91
410.82	01/24/90	LD-91-014	3/26/91
410.83	01/24/90	LD-91-014	3/26/91
410.84	01/24/90	LD-91-014	3/26/91
410.85	01/24/90	LD-91-014	3/26/91
410.86	01/24/90	LD-91-014	3/26/91
410.87	01/24/90	LD-91-014	3/26/91
410.88	01/24/90	LD-91-014	3/26/91
410.89	01/24/90	LD-91-014	3/26/91
410.90	01/24/90	LD-91-014	3/26/91
410.91	01/24/90	LD-91-014	3/26/91
410.92	01/24/90	LD-91-014	3/26/91
410.93	01/24/90	LD-91-014	3/26/91
410.94	01/24/90	LD-91-014	3/26/91
410.95	01/24/90	LD-91-014	3/26/91
410.96	01/24/90	LD-91-014	3/26/91
480.8	01/24/90	LD-92-024	2/18/92
480.9	01/24/90	LD-92-014	3/26/91
480.10	01/24/90	LD-92-014	3/26/91
480.11	01/24/90	LD-92-014	3/26/91
480.12	01/24/90	LD-92-014	3/26/91
480.13	01/24/90	LD-92-014	3/26/91
480.14	01/24/90	LD-92-014	3/26/91
480.15	01/24/90	LD-92-014	3/26/91
480.16	01/24/90	LD-92-014	3/26/91
480.17	01/24/90	LD-92-014	3/26/91

480.18	01/24/90	LD-92-014	3/26/91
480.19	01/24/90	LD-92-014	3/26/91
480.20	01/24/90	LD-92-014	3/26/91
480.21	01/24/90	LD-92-014	3/26/91
480.22	01/24/90	LD-92-014	3/26/91
480.23	01/24/90	LD-92-014	3/26/91
480.24	01/24/90	LD-92-014	3/26/91
480.25	01/24/90	LD-92-014	3/26/91
480.26	01/24/90	LD-92-014	3/26/91
480.27	01/24/90	LD-92-014	3/26/91
480.28	01/24/90	LD-92-014	3/26/91
480.29	01/24/90	LD-92-014	3/26/91
480.30	01/24/90	LD-92-014	3/26/91
480.31	01/24/90	LD-92-014	3/26/91
480.32	01/24/90	LD-92-014	3/26/91
480.33	01/24/90	LD-92-014	3/26/91
450.1	01/24/90	LD-92-014	3/26/91
450.2	01/24/90	LD-92-014	3/26/91
420.4	12/21/90	LD-91-016	4/12/91
420.5	12/21/90	LD-91-016	4/12/91
420.6	12/21/90	LD-91-016	4/12/91
420.7	12/21/90	LD-91-016	4/12/91
450.8	12/21/90	LD-91-016	4/12/91
420.9	12/21/90	LD-91-016	4/12/91
420.10	12/21/90	LD-91-016	4/12/91
420.11	12/21/90	LD-91-016	4/12/91
420.12	12/21/90	LD-91-016	4/12/91
450.13	12/21/90	LD-91-016	4/12/91
450.14	12/21/90	LD-91-016	4/12/91
420.15	12/21/90	LD-91-016	4/12/91
420.17	12/21/90	LD-91-016	4/12/91
420.18	12/21/90	LD-91-016	4/12/91
420.19	12/21/90	LD-91-016	4/12/91
450.20	12/21/90	LD-91-016	4/12/91
420.21	12/21/90	LD-91-016	4/12/91
420.22	12/21/90	LD-91-016	4/12/91
420.23	12/21/90	LD-91-016	4/12/91
420.24	12/21/90	LD-91-016	4/12/91
420.25	12/21/90	LD-91-016	4/12/91
450.26	12/21/90	LD-91-016	4/12/91
420.27	12/21/90	LD-91-016	4/12/91
420.28	12/21/90	LD-91-016	4/12/91
420.29	12/21/90	LD-91-016	4/12/91
420.30	12/21/90	LD-91-016	4/12/91
450.31	12/21/90	LD-91-016	4/12/91
450.32	12/21/90	LD-91-016	4/12/91
420.33	12/21/90	LD-91-016	4/12/91
420.34	12/21/90	LD-91-016	4/12/91
420.35	12/21/90	LD-91-016	4/12/91
420.36	12/21/90	LD-91-016	4/12/91
450.37	12/21/90	LD-91-016	4/12/91
420.38	12/21/90	LD-91-016	4/12/91
450.39	12/21/90	LD-91-016	4/12/91
420.40	12/21/90	LD-91-016	4/12/91
420.41	12/21/90	LD-91-016	4/12/91

420.42	12/21/90	LD-91-016	4/12/91
420.43	12/21/90	LD-91-016	4/12/91
420.44	12/21/90	LD-91-016	4/12/91
450.45	12/21/90	LD-91-016	4/12/91
420.46	12/21/90	LD-91-016	4/12/91
420.47	12/21/90	LD-91-016	4/12/91
420.48	12/21/90	LD-91-016	4/12/91
420.49	12/21/90	LD-91-016	4/12/91
450.50	12/21/90	LD-91-016	4/12/91
450.51	12/21/90	LD-91-016	4/12/91
420.52	12/21/90	LD-91-016	4/12/91
420.53	12/21/90	LD-91-016	4/12/91
420.54	12/21/90	LD-91-016	4/12/91
420.55	12/21/90	LD-91-016	4/12/91
450.56	12/21/90	LD-91-016	4/12/91
420.57	12/21/90	LD-91-016	4/12/91
420.58	12/21/90	LD-91-016	4/12/91
450.59	12/21/90	LD-91-016	4/12/91
420.60	12/21/90	LD-91-016	4/12/91
100.1	12/21/90	LD-91-016	4/12/91
620.1	12/21/90	LD-91-016	4/12/91
620.2	12/21/90	LD-91-016	4/12/91
620.3	12/21/90	LD-91-016	4/12/91
620.4	12/21/90	LD-91-016	4/12/91
620.5	12/21/90	LD-91-016	4/12/91
620.6	12/21/90	LD-91-016	4/12/91
620.7	12/21/90	LD-91-016	4/12/91
620.8	12/21/90	LD-91-016	4/12/91
620.9	12/21/90	LD-91-016	4/12/91
620.10	12/21/90	LD-91-016	4/12/91
620.11	12/21/90	LD-91-016	4/12/91
620.12	12/21/90	LD-91-016	4/12/91
620.13	12/21/90	LD-91-016	4/12/91
620.14	12/21/90	LD-91-016	4/12/91
620.15	12/21/90	LD-91-016	4/12/91
620.16	12/21/90	LD-91-016	4/12/91
620.17	12/21/90	LD-91-016	4/12/91
620.18	12/21/90	LD-91-016	4/12/91
620.19	12/21/90	LD-91-016	4/12/91
620.20	12/21/90	LD-91-016	4/12/91
620.21	12/21/90	LD-91-016	4/12/91
620.22	12/21/90	LD-91-016	4/12/91
620.23	12/21/90	LD-91-016	4/12/91
620.24	12/21/90	LD-91-016	4/12/91
620.25	12/21/90	LD-91-016	4/12/91
620.26	12/21/90	LD-91-016	4/12/91
620.27	12/21/90	LD-91-016	4/12/91
620.28	12/21/90	LD-91-016	4/12/91
620.28	12/21/90	LD-91-016	4/12/91
620.30	12/21/90	LD-91-016	4/12/91
620.31	12/21/90	LD-91-016	4/12/91
620.32	12/21/90	LD-91-016	4/12/91
620.33	12/21/90	LD-91-016	4/12/91
620.34	12/21/90	LD-91-016	4/12/91
620.35	12/21/90	LD-91-016	4/12/91

620.36	12/21/90	LD-91-016	4/12/91
620.37	12/21/90	LD-91-016	4/12/91
620.38	12/21/90	LD-91-016	4/12/91
440.10	12/24/90	LD-91-018	4/26/91
440.11	12/24/90	LD-91-018	4/26/91
440.12	12/24/90	LD-91-018	4/26/91
440.13	12/24/90	LD-91-018	4/26/91
440.14	12/24/90	LD-91-018	4/26/91
440.15	12/24/90	LD-91-018	4/26/91
440.16a	12/24/90	LD-92-008	1/29/92
440.16b	12/24/90	LD-92-008	1/29/92
440.16c	12/24/90	LD-92-008	1/29/92
440.16d	12/24/90	LD-92-008	1/29/92
440.16e	12/24/90	LD-91-018	4/26/91
440.16f	12/24/90	LD-92-008	1/29/92
440.16g	12/24/90	LD-92-008	1/29/92
440.16h	12/24/90	LD-92-008	1/29/92
440.16i	12/24/90	LD-92-008	1/29/92
440.16j	12/24/90	LD-92-008	1/29/92
440.17	12/24/90	LD-91-018	4/26/91
440.18	12/24/90	LD-91-018	4/26/91
440.19	12/24/90	LD-91-018	4/26/91
440.20	12/24/90	LD-91-018	4/26/91
440.21	12/24/90	LD-91-018	4/26/91
440.22	12/24/90	LD-91-018	4/26/91
440.23	12/24/90	LD-92-024	2/18/92
440.24	12/24/90	LD-91-018	4/26/91
440.25	12/24/90	LD-91-018	4/26/91
440.26	12/24/90	LD-91-018	4/26/91
440.27	12/24/90	LD-91-018	4/26/91
440.28	12/24/90	LD-91-018	4/26/91
440.29	12/24/90	LD-91-018	4/26/91
440.30	12/24/90	LD-91-018	4/26/91
440.31	12/24/90	LD-91-018	4/26/91
440.32	01/31/91	LD-91-019	5/6/91
440.33	01/31/91	LD-91-019	5/6/91
440.34	01/31/91	LD-91-019	5/6/91
440.35	01/31/91	LD-92-008	1/29/92
440.36	01/31/91	LD-91-019	5/16/91
440.37	01/31/91	LD-91-019	5/16/91
440.38	01/31/91	LD-91-019	5/16/91
440.39	02/15/91	LD-91-071	12/24/91
440.40	02/15/91	LD-91-024	5/16/91
440.41	02/15/91	LD-92-020	2/14/92
440.42	02/15/91	LD-92-020	2/14/92
440.43	02/15/91	LD-91-024	05/16/91
440.44	02/15/91	LD-92-020	2/14/92
440.45	02/15/91	LD-92-020	2/14/92
440.46	02/15/91	LD-91-024	5/16/91
440.47	02/15/91	LD-91-024	5/16/91
440.48	02/15/91	LD-91-071	12/24/91
440.49	02/15/91	LD-91-024	5/15/91
440.50	02/15/91	LD-91-024	5/15/91
440.51	02/15/91	LD-91-024	5/15/91
440.52	02/15/91	LD-92-020	2/14/92

440.53	02/15/91	LD-91-071	12/24/91
440.54	02/15/91	LD-91-024	5/16/91
440.55	02/15/91	LD-91-024	5/16/91
440.56	02/15/91	LD-91-024	5/16/91
440.57	02/15/91	LD-91-071	12/24/91
440.58	02/15/91	LD-91-071	12/24/91
440.59	02/15/91	LD-91-071	12/24/91
440.60	02/15/91	LD-91-071	12/24/91
440.61	02/15/91	LD-91-071	12/24/91
440.62	02/15/91	LD-91-071	12/24/91
440.63	02/15/91	LD-91-071	12/24/91
440.64	02/15/91	LD-91-024	5/16/91
440.65	02/15/91	LD-91-071	12/24/91
440.66	02/15/91	LD-91-071	12/24/91
440.67	02/15/91	LD-91-071	12/24/91
440.68	02/15/91	LD-91-024	5/16/91
440.69	02/15/91	LD-91-024	5/16/91
440.70	02/15/91	LD-91-071	12/24/91
440.71	02/15/91	LD-91-071	12/24/91
440.72	02/15/91	LD-92-020	2/14/92
440.73	02/15/91	LD-92-020	2/14/92
440.74	02/15/91	LD-91-024	5/16/91
440.75	02/15/91	LD-91-071	12/24/91
440.76	02/15/91	LD-91-024	5/16/91
440.77	02/15/91	LD-91-024	5/16/91
440.78	02/15/91	LD-91-024	5/16/91
440.79	02/15/91	LD-91-024	5/16/91
440.80	02/15/91	LD-91-024	5/16/91
440.81	02/15/91	LD-91-024	5/16/91
440.82	02/15/91	LD-92-020	2/14/92
	02/21/91	LD-91-069	12/23/91
	04/12/91	LD-91-021	5/13/91
440.83	05/13/91	LD-91-071	12/24/91
440.84	05/13/91	LD-91-062	11/27/91
440.85	05/13/91	LD-91-062	11/27/91
440.85a	05/13/91	LD-91-071	12/24/91
440.85b	05/13/91	LD-91-071	12/24/91
440.85c	05/13/91	LD-91-071	12/24/91
440.86d	05/13/91	LD-91-062	11/27/91
440.86e	05/13/91	LD-91-062	11/27/91
440.86f	05/13/91	LD-92-008	1/29/92
440.87	05/13/91	LD-91-062	11/27/91
440.88	05/13/91	LD-91-062	11/27/91
440.89	05/13/91	LD-91-062	11/27/91
450.01	05/13/91	LD-91-062	11/27/91
440.90	05/13/91	LD-91-062	11/27/91
440.91	05/13/91	LD-92-008	1/29/92
440.92	05/13/91	LD-91-061	11/27/91
440.93	05/13/91	LD-91-061	11/27/91
440.94	05/13/91	LD-91-061	11/27/91
450.02	05/13/91	LD-91-061	11/27/91
440.95a	05/13/91	LD-91-061	11/27/91
440.95b	05/13/91	LD-91-061	11/27/91
440.96	05/13/91	LD-91-061	11/27/91

450.03	05/13/91	LD-91-061	11/27/91
440.97	05/13/91	LD-91-071	12/24/91
440.98	05/13/91	LD-91-062	11/27/91
440.99	05/13/91	LD-91-062	11/27/91
440.100	05/13/91	LD-91-071	12/24/91
440.101	05/13/91	LD-91-062	11/27/91
440.102	05/13/91	LD-91-062	11/27/91
440.103	05/13/91	LD-91-062	11/27/91
440.104	05/13/91	LD-91-071	12/24/91
440.105	05/13/91	LD-92-024	2/18/92
450.04	05/13/91	LD-92-024	2/18/92
450.05	05/13/91	LD-91-071	12/24/91
440.106	05/13/91	LD-91-071	12/24/91
440.107	05/13/91	LD-91-062	11/27/91
450.06	05/13/91	LD-91-062	11/27/91
440.108	05/13/91	LD-91-071	12/24/91
450.07	05/13/91	LD-91-071	12/24/91
440.109	05/13/91	LD-91-062	11/27/91
440.110(1)	05/13/91	LD-92-020	2/14/92
440.110(2)	05/13/91	LD-92-020	2/14/92
440.110(3)	05/13/91	LD-92-020	2/14/92
440.110(4)	05/13/91	LD-92-020	2/14/92
440.110(5)	05/13/91	LD-92-020	2/14/92
440.110(6)	05/13/91	LD-92-020	2/14/92
440.111	05/13/91	LD-92-020	2/14/92
	07/22/91	LD-91-063	12/2/91
430.1	07/29/91	LD-91-067	12/20/91
430.2	07/29/91	LD-91-067	12/20/91
430.3	07/29/91	LD-91-067	12/20/91
430.4	07/29/91	LD-91-067	12/20/91
430.5	07/29/91	LD-91-067	12/20/91
430.6	07/29/91	LD-92-001	1/14/92
430.7	07/29/91	LD-92-001	1/14/92
430.8	07/29/91	LD-92-001	1/14/92
430.9	07/29/91	LD-92-024	2/18/92
430.10	07/29/91	LD-92-001	1/13/92
430.11	07/29/91	LD-91-067	12/20/91
430.12	07/29/91	LD-92-001	1/14/92
430.13	07/29/91	LD-92-001	1/14/92
430.14	07/29/91	LD-91-067	12/20/91
430.15	07/29/91	LD-92-024	2/18/92
430.16	07/29/91	LD-92-001	1/14/92
430.17	07/29/91	LD-91-067	12/20/91
430.18	07/29/91	LD-91-067	12/20/91
430.19	07/29/91	LD-91-067	12/20/91
430.20	07/29/91	LD-91-067	12/20/91
430.21	07/29/91	LD-91-067	12/20/91
430.22	07/29/91	LD-91-067	12/20/91
430.23	07/29/91	LD-92-001	1/14/92
430.24	07/29/91	LD-91-067	12/20/91
430.25	07/29/91	LD-92-024	2/18/92
430.26	07/29/91	LD-92-024	2/18/92
430.27	07/29/91	LD-91-067	12/20/91
430.28	07/29/91	LD-91-067	12/20/91
430.29	07/29/91	LD-91-067	12/20/91

430.30	07/29/91	LD-91-067	12/20/91
430.31	07/29/91	LD-91-067	12/20/91
430.32	07/29/91	LD-91-067	12/20/91
430.33	07/29/91	LD-91-067	12/20/91
430.34	07/29/91	LD-91-067	12/20/91
430.35	07/29/91	LD-91-067	12/20/91
430.36	07/29/91	LD-91-067	12/20/91
430.37	07/29/91	LD-91-067	12/20/91
430.38	07/29/91	LD-91-067	12/20/91
430.39	07/29/91	LD-91-067	12/20/91
430.40	07/29/91	LD-91-067	12/20/91
430.41	07/29/91	LD-91-067	12/20/91
430.42	07/29/91	LD-91-067	12/20/91
430.43	07/29/91	LD-91-067	12/20/91
430.44	07/29/91	LD-91-067	12/20/91
430.45	07/29/91	LD-91-067	12/20/91
430.46	07/29/91	LD-91-067	12/20/91
430.47	07/29/91	LD-91-067	12/20/91
430.48	07/29/91	LD-91-067	12/20/91
430.49	07/29/91	LD-92-024	2/18/92
430.50	07/29/91	LD-92-024	2/18/92
430.51	07/29/91	LD-91-067	12/20/91
	08/02/91	LD-91-047	8/9/91
		LD-91-048	8/14/91
471.1	08/03/91	LD-91-065	12/13/19
471.2	08/03/91	LD-91-065	12/13/19
471.3	08/03/91	LD-91-065	12/13/19
471.4	08/03/91	LD-91-065	12/13/19
471.5	08/03/91	LD-92-024	2/18/92
471.6	08/03/91	LD-91-065	12/13/91
471.7	08/03/91	LD-91-065	12/13/91
471.8	08/03/91	LD-91-065	12/13/91
471.9	08/03/91	LD-91-065	12/13/91
471.10	08/03/91	LD-91-065	12/24/91
471.11	08/03/91	LD-91-065	12/13/91
471.12	08/03/91	LD-91-065	12/13/91
471.13	08/03/91	LD-91-065	12/13/91
471.14	08/03/91	LD-91-070	12/24/91
471.15	08/03/91	LD-91-065	12/13/91
471.16	08/03/91	LD-91-065	12/13/91
471.17	08/03/91	LD-91-065	12/13/91
471.18	08/03/91	LD-91-065	12/13/91
471.19	08/03/91	LD-91-065	12/13/91
471.20	08/03/91	LD-91-065	12/13/91
471.21	08/03/91	LD-91-065	12/13/91
471.22	08/03/91	LD-91-070	12/24/91
471.23	08/03/91	LD-91-065	12/13/91
471.24	08/03/91	LD-91-065	12/13/91
471.25	08/03/91	LD-92-024	2/18/92
471.26	08/03/91	LD-91-065	12/13/91
471.27	08/03/91	LD-91-065	12/13/91
471.28	08/03/91	LD-91-065	12/13/91
471.29	08/03/91	LD-91-065	12/13/91
471.30	08/03/91	LD-91-065	12/13/91
471.31	08/03/91	LD-91-065	12/13/91

471.32	08/03/91	LD-91-065	12/13/91
471.33	08/03/91	LD-91-065	12/13/91
471.34	08/03/91	LD-91-065	12/13/91
471.35	08/03/91	LD-91-065	12/13/91
471.36	08/03/91	LD-91-065	12/13/91
471.37	08/03/91	LD-91-065	12/13/91
471.38	08/03/91	LD-91-065	12/13/91
450.8	08/03/91	LD-91-061	11/27/91
450.9	08/03/91	LD-91-071	12/24/91
450.10	08/03/91	LD-91-061	11/27/91
450.11	08/03/91	LD-91-061	11/27/91
450.12	08/03/91	LD-91-071	12/24/91
450.13	08/03/91	LD-91-071	12/24/91
450.14	08/03/91	LD-91-071	12/24/91
450.15	08/03/91	LD-91-071	12/24/91
500.15	08/05/91	LD-91-066	12/17/91
500.16(a)	08/05/91	LD-91-066	12/17/91
500.16(b)	08/05/91	LD-91-066	12/17/91
500.16(c)	08/05/91	LD-91-066	12/17/91
500.17(a)	08/05/91	LD-91-066	12/17/91
500.1(b)	08/05/91	LD-91-066	12/17/91
500.18	08/05/91	LD-91-066	12/17/91
500.19	08/05/91	LD-91-066	12/17/91
500.20	08/05/91	LD-91-066	12/17/91
500.21	08/05/91	LD-91-066	12/17/91
500.22	08/05/91	LD-91-066	12/17/91
500.23	08/05/91	LD-91-066	12/17/91
500.24(a)	08/05/91	LD-92-024	2/18/92
500.24(b)	08/05/91	LD-92-024	2/18/92
500.25(a)	08/05/91	LD-92-024	2/18/92
500.25(b)	08/05/91	LD-92-024	2/18/92
500.26	08/05/91	LD-91-066	12/17/91
500.27	08/05/91	LD-91-066	12/17/91
500.28	08/05/91	LD-91-066	12/17/91
500.29(a)	08/05/91	LD-91-066	12/17/91
500.30	08/05/91	LD-91-066	12/17/91
500.31	08/05/91	LD-91-066	12/17/91
500.32(a)	08/05/91	LD-91-066	12/17/91
500.32(b)	08/05/91	LD-91-066	12/17/91
500.32(c)	08/05/91	LD-91-066	12/17/91
500.33	08/05/91	LD-91-066	12/17/91
500.34	08/05/91	LD-91-066	12/17/91
500.35	08/05/91	LD-91-066	12/17/91
260.23	08/06/91	LD-92-021	2/18/92
260.24(a)	08/06/91	LD-92-021	2/18/92
260.24(b)	08/06/91	LD-92-021	2/18/92
260.24(c)	08/06/91	LD-92-021	2/18/92
260.24(d)	08/06/91	LD-92-021	2/18/92
260.24(e)	08/06/91	LD-92-021	2/18/92
260.25	08/06/91	LD-92-021	2/18/92
260.26	08/06/91	LD-92-021	2/18/92
260.27(1)	08/06/91	LD-92-021	2/18/92
260.27(2)	08/06/91	LD-92-021	2/18/92
260.28	08/06/91	LD-92-021	2/18/92
252.02	08/08/91	LD-92-016	2/12/92

252.03	08/08/91	LD-92-007	1/24/92
252.04	08/08/91	LD-91-071	12/24/91
252.05	08/08/91	LD-91-071	12/24/91
252.06	08/08/91	LD-91-071	12/24/91
252.07	08/08/91	LD-92-009	1/29/92
252.08	08/08/91	LD-92-009	1/29/92
252.09	08/08/91	LD-92-024	2/18/92
252.10	08/08/91	LD-92-024	2/18/92
252.11	08/08/91	LD-91-071	12/24/91
252.12	08/08/91	LD-91-071	12/24/91
252.13	08/08/91	LD-91-071	12/24/91
252.14	08/08/91	LD-91-071	12/24/91
252.15	08/08/91	LD-92-007	1/24/92
281.35	08/08/91	LD-91-071	12/24/91
281.36	08/08/91	LD-91-071	12/24/91
281.37	08/08/91	LD-91-060	11/27/91
281.38	08/08/91	LD-91-060	11/27/91
281.39	08/08/91	LD-91-060	11/27/91
281.40	08/08/91	LD-91-060	11/27/91
281.41	08/08/91	LD-92-009	1/29/92
281.42	08/08/91	LD-91-071	12/24/91
281.43	08/08/91	LD-91-071	12/24/91
281.44	08/08/91	LD-91-071	12/24/91
281.45	08/08/91	LD-92-024	2/18/92
281.46	08/08/91	LD-91-060	11/27/91
281.47	08/08/91	LD-91-060	11/27/91
281.48	08/08/91	LD-91-071	12/24/91
281.49	08/08/91	LD-91-071	12/24/91
281.50*	08/08/91	LD-91-071	12/24/91
440.113	08/08/91	LD-91-071	12/24/91
440.114	08/08/91	LD-91-071	12/24/91
440.115	08/21/91	LD-92-020	2/14/92
		LD-92-024	2/18/92
440.116	08/21/91	LD-92-020	2/14/92
440.117	08/21/91	LD-92-020	2/14/92
440.118	08/21/91	LD-92-020	2/14/92
440.119(1)	08/21/91	LD-92-020	2/14/92
440.119(2)	08/21/91	LD-92-020	2/14/92
440.119(3)	08/21/91	LD-92-020	2/14/92
440.120	08/21/91	LD-92-020	2/14/92
440.121	08/21/91	LD-92-020	2/14/92
440.122	08/21/91	LD-92-020	2/14/92
440.123	08/21/91	LD-92-020	2/14/92
440.124	08/21/91	LD-92-020	2/14/92
440.125	08/21/91	LD-92-020	2/14/92
440.126	08/21/91	LD-92-020	2/14/92
440.17(1)	08/21/91	LD-92-020	2/14/92
440.128	08/21/91	LD-91-071	12/24/91
440.129	08/21/91	LD-92-008	1/29/92
440.130	08/21/91	LD-92-008	1/29/92
440.131	08/21/91	LD-92-008	1/29/92
440.132	08/21/91	LD-92-008	1/29/92
440.133	08/21/91	LD-92-008	1/29/92
440.134	08/21/91	LD-92-008	1/29/92
		LD-92-020	2/14/92

440.135	08/21/91	LD-92-008	1/29/92
440.136	08/21/91	LD-92-020	2/14/91
440.137	08/21/91	LD-92-008	1/29/92
440.138	08/21/91	LD-92-008	1/29/92
		LD-92-020	2/14/92
440.139	08/21/91	LD-91-062	11/27/91
440.140	08/21/91	LD-92-008	1/29/92
440.141	08/21/91	LD-92-008	1/29/92
440.142	08/21/91	LD-92-008	1/29/92
440.143	08/21/91	LD-92-008	1/29/92
440.144	08/21/91	LD-92-008	1/29/92
440.145	08/21/91	LD-92-008	1/29/92
440.146	08/21/91	LD-92-008	1/29/92
440.147	08/21/91	LD-92-008	1/29/92
440.148	08/21/91	LD-92-008	1/29/92
440.149	08/21/91	LD-92-008	1/29/92
440.150	08/21/91	LD-92-008	1/29/92
440.151	08/21/91	LD-92-008	1/29/92
	09/16/91	LD-91-054	10/22/91
1	09/19/91	LD-92-022	2/18/92
2	09/19/91	LD-92-022	2/18/92
3	09/19/91	LD-92-022	2/18/92
4	09/19/91	LD-92-022	2/18/92
5	09/19/91	LD-92-022	2/18/92
6	09/19/91	LD-92-022	2/18/92
7	09/19/91	LD-92-022	2/18/92
8	09/19/91	LD-92-022	2/18/92
9	09/19/91	LD-92-022	2/18/92
10	09/19/91	LD-92-022	2/18/92
11	09/19/91	LD-92-022	2/18/92
12	09/19/91	LD-92-022	2/18/92
13	09/19/91	LD-92-022	2/18/92
14	09/19/91	LD-92-022	2/18/92
620.6A	09/19/91		
620.7A	09/19/91		
620.8A	09/19/91		
620.9A	09/19/91		
620.10A	09/19/91		
420.61	09/25/91	LD-91-064	12/9/91
420.62	09/25/91	LD-92-024	2/18/92
420.63	09/25/91	LD-92-024	2/18/92
420.64	09/25/91	LD-92-024	2/18/92
210.1	09/26/91	LD-92-016	2/12/92
210.2	09/26/91	LD-92-016	2/12/92
210.3	09/26/91	LD-92-024	2/18/92
210.4	09/26/91	LD-92-016	2/12/92
210.5	09/26/91	LD-92-016	2/12/92
210.6	09/26/91	LD-92-016	2/12/92
210.7	09/26/91	LD-92-016	2/12/92
210.8	09/26/91	LD-92-007	1/24/92
210.9	09/26/91	LD-92-016	2/12/92
210.10	09/26/91	LD-92-016	2/12/92
210.11	09/26/91	LD-92-016	2/12/92
210.12	09/26/91	LD-92-007	1/24/92
210.13	09/26/91	LD-92-007	1/24/92

210.14	09/26/91	LD-92-007	1/24/92
210.15	09/26/91	LD-92-007	1/24/92
210.16	09/26/91	LD-92-007	1/24/92
210.17	09/26/91	LD-92-007	1/24/92
210.18	09/26/91	LD-92-007	1/24/92
210.19	09/26/91	LD-92-007	1/24/92
210.20	09/26/91	LD-92-007	1/24/92
210.21	09/26/91	LD-92-007	1/24/92
210.22	09/26/91	LD-92-007	1/24/92
210.23	09/26/91	LD-92-007	1/24/92
210.24	09/26/91	LD-92-007	1/24/92
210.25	09/26/91	LD-92-007	1/24/92
210.26	09/26/91	LD-92-007	1/24/92
210.27	09/26/91	LD-92-007	1/24/92
210.28	09/26/91	LD-92-007	1/24/92
210.29	09/26/91	LD-92-007	1/24/92
210.30	09/26/91	LD-92-007	1/24/92
210.31	09/26/91	LD-92-007	1/24/92
210.32	09/26/91	LD-92-007	1/24/92
210.33	09/26/91	LD-92-016	2/12/92
210.34	09/26/91	LD-92-016	2/12/92
210.35	09/26/91	LD-92-016	2/12/92
210.36	09/26/91	LD-92-016	2/12/92
210.37	09/26/91	LD-92-007	1/24/92
210.38	09/26/91	LD-92-007	1/24/92
210.39	09/26/91	LD-92-016	2/12/92
210.40	09/26/91	LD-92-007	1/24/92
210.41	09/26/91	LD-92-007	1/24/92
210.42	09/26/91	LD-92-016	2/12/92
210.43	09/26/91	LD-92-016	2/12/92
210.44	09/26/91	LD-92-007	1/24/92
210.45	09/26/91	LD-92-016	2/12/92
210.46	09/26/91	LD-92-016	2/12/92
210.47	09/26/91	LD-92-016	2/12/92
210.48	09/26/91	LD-92-016	2/12/92
210.49	09/26/91	LD-92-007	1/24/92
210.50	09/26/91	LD-92-016	2/12/92
210.51	09/26/91	LD-92-016	2/12/92
210.52	09/26/91	LD-92-016	2/12/92
210.53	09/26/91	LD-92-016	2/12/92
210.54	09/26/91	LD-92-007	1/24/92
210.55	09/26/91	LD-92-007	1/24/92
210.56	09/26/91	LD-92-007	1/24/92
210.57	09/26/91	LD-92-016	2/12/92
210.58	09/26/91	LD-92-016	2/12/92
210.59	09/26/91	LD-92-016	2/12/92
210.60	09/26/91	LD-92-016	2/12/92
210.61	09/26/91	LD-92-016	2/12/92
210.62	09/26/91	LD-92-016	2/12/92
210.63	09/26/91	LD-92-016	2/12/92
210.64	09/26/91	LD-92-016	2/12/92
210.65	09/26/91	LD-92-016	2/12/92
210.66	09/26/91	LD-92-016	2/12/92
210.67	09/26/91	LD-92-016	2/12/92
210.68	09/26/91	LD-92-016	2/12/92

210.69	09/26/91	LD-92-016	2/12/92
210.70	09/26/91	LD-92-016	2/12/92
210.71	09/26/91	LD-92-016	2/12/92
210.72	09/26/91	LD-92-016	2/12/92
210.73	09/26/91	LD-92-016	2/12/92
210.74	09/26/91	LD-92-007	1/24/92
210.75	09/26/91	LD-92-007	1/24/92
210.76	09/26/91	LD-92-016	2/12/92
210.77	09/26/91	LD-92-016	2/12/92
210.78	09/26/91	LD-92-016	2/12/92
210.79	09/26/91	LD-92-016	2/12/92
210.80	09/26/91	LD-92-016	2/12/92
210.81	09/26/91	LD-92-016	2/12/92
210.82	09/26/91	LD-92-016	2/12/92
210.83	09/26/91	LD-92-016	2/12/92
210.84	09/26/91	LD-92-016	2/12/92
210.85	09/26/91	LD-92-016	2/12/92
210.86	09/26/91	LD-92-016	2/12/92
210.87	09/26/91	LD-92-016	2/12/92
210.88	09/26/91	LD-92-024	2/18/92
210.89	09/26/91	LD-92-009	1/29/92
210.90	09/26/91	LD-92-007	1/24/92
210.91	09/26/91	LD-92-007	1/24/92
210.92	09/26/91	LD-92-007	1/24/92
210.93	09/26/91	LD-92-007	1/24/92
210.94	09/26/91	LD-92-016	
210.95	09/26/91		
210.96	09/26/91		
220.52	09/26/91		
220.53	09/26/91		
220.54	09/26/91		
220.55	09/26/91	LD-92-024	2/18/92
220.56	09/26/91	LD-92-016	2/12/92
252.16	09/26/91	LD-92-016	2/12/92
252.17	09/26/91	LD-92-016	2/12/92
252.18	09/26/91	LD-92-016	2/12/92
252.19	09/26/91	LD-92-016	2/12/92
252.20	09/26/91	LD-92-016	2/12/92
311.1	09/26/91	LD-92-016	2/12/92
311.2	09/26/91	LD-92-016	2/12/92
230.1	09/26/91		
230.2	09/26/91	LD-92-016	2/12/92
230.3	09/26/91	LD-92-016	2/12/92
230.4	09/26/91	LD-92-024	2/18/92
230.5	09/26/91	LD-92-016	2/12/92
230.6	09/26/91	Submitted 2/25/92	
230.7	09/26/91	LD-92-016	2/12/92
230.8	09/26/91	Submitted 2/25/92	
230.9	09/26/91	LD-92-016	2/12/92
230.10	09/26/91	LD-92-024	2/18/92
220.0	09/26/91	LD-92-016	2/12/92
220.1	09/26/91	LD-92-016	2/12/92
220.2	09/26/91	LD-92-016	2/12/92
220.3	09/26/91	LD-92-016	2/12/92
220.4	09/26/91	LD-92-016	2/12/92

220.5	09/26/91	LD-92-016	2/12/92
220.6	09/26/91	LD-92-024	2/18/92
220.7	09/26/91	LD-92-024	2/18/92
220.8	09/26/91	LD-92-016	2/12/92
220.9	09/26/91	Submitted 2/25/92	
220.10	09/26/91	LD-92-016	2/12/92
220.11	09/26/91	LD-92-016	2/12/92
220.12	09/26/91	LD-92-016	2/12/92
220.13	09/26/91	LD-92-016	2/12/92
220.14	09/26/91	LD-92-016	2/12/92
220.15	09/26/91	LD-92-016	2/12/92
220.16	09/26/91	LD-92-016	2/12/92
220.17	09/26/91	LD-92-016	2/12/92
220.18	09/26/91	LD-92-016	2/12/92
220.19	09/26/91	LD-92-016	2/12/92
220.20	09/26/91	LD-92-016	2/12/92
220.21	09/26/91	LD-92-016	2/12/92
220.22	09/26/91	LD-92-016	2/12/92
220.23	09/26/91	LD-92-016	2/12/92
220.24	09/26/91	LD-92-016	2/12/92
220.25	09/26/91	LD-92-016	2/12/92
220.26	09/26/91	LD-92-016	2/12/92
220.27	09/26/91	LD-92-016	2/12/92
220.28	09/26/91	LD-92-016	2/12/92
220.29	09/26/91	LD-92-016	2/12/92
220.30	09/26/91	LD-92-016	2/12/92
220.31	09/26/91	LD-92-016	2/12/92
220.32	09/26/91	LD-92-016	2/12/92
220.33	09/26/91	LD-92-016	2/12/92
220.34	09/26/91	LD-92-016	2/12/92
220.35	09/26/91	LD-92-016	2/12/92
220.36	09/26/91	LD-92-016	2/12/92
220.37	09/26/91	LD-92-016	2/12/92
220.38	09/26/91	LD-92-016	2/12/92
220.39	09/26/91	LD-92-016	2/12/92
220.40	09/26/91	LD-92-016	2/12/92
220.41	09/26/91	LD-92-016	2/12/92
270.42	09/26/91	LD-92-016	2/12/92
270.43	09/26/91	LD-92-016	2/12/92
270.44	09/26/91	LD-92-016	2/12/92
220.45	09/26/91	LD-92-016	2/12/92
220.46	09/26/91	LD-92-016	2/12/92
220.47	09/26/91	LD-92-016	2/12/92
220.48	09/26/91	LD-92-016	2/12/92
220.49	09/26/91	LD-92-016	2/12/92
220.50	09/26/91	LD-92-007	1/24/92
220.51	09/26/91	LD-92-016	2/12/92
810.1	10/09/91	LD-92-024	2/18/92
810.2(a)	10/09/91	LD-92-009	1/29/92
810.2(b)	10/09/91	LD-92-009	1/29/92
810.2(c)	10/09/91	LD-92-009	1/29/92
810.3	10/09/91	LD-92-009	1/29/92
100.2	10/09/91	LD-92-024	2/18/92
640.1	10/10/91	LD-92-021	2/18/92
640.2	10/10/91	LD-92-021	2/18/92

640.3	10/10/91	LD-92-021	2/18/92
640.4	10/10/91	LD-92-021	2/18/92
640.5	10/10/91	LD-92-021	2/18/92
640.6	10/10/91	LD-92-021	2/18/92
640.7	10/10/91	LD-92-021	2/18/92
640.8	10/10/91	LD-92-021	2/18/92
640.9	10/10/91	LD-92-021	2/18/92
640.10	10/10/91	LD-92-021	2/18/92
640.11	10/10/91	LD-92-021	2/18/92
640.12	10/10/91	LD-92-021	2/18/92
1	10/10/91	LD-92-010	1/31/92
2	10/10/91	LD-92-010	1/31/92
3	10/10/91	LD-92-010	1/31/92
4	10/10/91	LD-92-010	1/31/92
5	10/10/91	LD-92-010	1/31/92
6	10/10/91	LD-92-010	1/31/92
7	10/10/91	LD-92-010	1/31/92
8	10/10/91	LD-92-010	1/31/92
1	10/16/91	LD-92-024	2/18/92
2	10/16/91	LD-92-024	2/18/92
410.32.c	10/10/91	LD-92-006	1/24/92
410.32.g	10/10/91	LD-92-006	1/24/92
410.97	10/10/91	LD-92-006	1/24/92
410.98	10/10/91	LD-92-006	1/24/92
410.33	10/10/91	LD-92-006	1/24/92
410.36	10/10/91	LD-92-006	1/24/92
410.99	10/10/91	LD-92-006	1/24/92
410.100	10/10/91	LD-92-006	1/24/92
480.5	10/10/91	LD-92-006	1/24/92
480.34	10/10/91	LD-92-006	1/24/92
270.1	10/10/91	LD-92-017	2/12/92
270.2a	10/10/91	LD-92-017	2/12/92
270.2b	10/10/91	LD-92-017	2/12/92
270.2c	10/10/91	LD-92-017	2/12/92
270.2d	10/10/91	LD-92-017	2/12/92
2.0.2e	10/10/91	LD-92-017	2/12/92
270.2f	10/10/91	LD-92-017	2/12/92
410.101	10/10/91	LD-92-017	2/12/92
410.102a	10/10/91	LD-92-017	2/12/92
410.102b	10/10/91	LD-92-017	2/12/92
480.35	10/10/91	LD-92-024	2/18/92
480.35a	10/10/91	LD-92-017	2/12/92
480.35b	10/10/91	LD-92-024	2/18/92
480.35c	10/10/91	LD-92-017	2/12/92
480.35d	10/10/91	LD-92-024	2/18/92
480.35e	10/10/91	LD-92-017	2/12/92
480.36a	10/10/91	LD-92-006	1/24/92
480.36b	10/10/91	LD-92-006	1/24/92
480.36c	10/10/91	LD-92-006	1/24/92
480.36d	10/10/91	LD-92-006	1/24/92
480.37a	10/10/91	LD-92-006	1/24/92
480.37b	10/10/91	LD-92-006	1/24/92
480.37c	10/10/91	LD-92-006	1/24/92
480.37d	10/10/91	LD-92-071	12/24/91
450.3a	10/10/91	LD-92-006	1/24/92

450.3b	10/10/91	LD-92-006	1/24/92
450.3c	10/10/91	LD-92-006	1/24/92
450.3d	10/10/91	LD-92-006	1/24/92
450.3e	10/10/91	LD-92-006	1/24/92
450.3f	10/10/91	LD-92-006	1/24/92
450.3g	10/10/91	LD-92-006	1/24/92
410.103a	10/10/91	LD-92-017	2/12/92
410.103b	10/10/91	LD-92-017	2/12/92
410.103c	10/10/91	LD-92-017	2/12/92
410.103d	10/10/91	LD-92-017	2/12/92
410.103e	10/10/91	LD-92-017	2/12/92
410.103f	10/10/91	LD-92-017	2/12/92
410.103g	10/10/91	LD-92-008	1/29/92
		LD-92-017	2/12/92
410.103h	10/10/91	LD-92-017	2/12/92
410.103i	10/10/91	LD-92-017	2/12/92
410.64a	10/10/91	LD-92-017	2/12/92
410.64b	10/10/91	LD-92-017	2/12/92
410.54	10/10/91	LD-92-017	2/12/92
410.104a	10/10/91	LD-92-017	2/12/92
410.104b	10/10/91	LD-92-017	2/12/92
410.10c	10/10/91	LD-92-017	2/12/92
410.10d	10/10/91	LD-92-008	1/29/92
410.67	10/10/91	LD-92-017	2/12/92
410.61	10/10/91	LD-92-017	2/12/92
410.68	10/10/91	LD-92-017	2/12/92
410.56	10/10/91	LD-92-017	2/12/92
410.59	10/10/91	LD-92-017	2/12/92
410.55a	10/10/91	LD-92-017	2/12/92
410.55d	10/10/91	LD-92-017	2/12/92
410.105	10/10/91	LD-92-017	2/12/92
281.34	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.106a	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.106b	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.107a	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.107b	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.107c	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.107d	10/10/91	LD-92-017	2/12/92
		LD-92-024	2/18/92
410.108	10/10/91	LD-92-017	2/12/92
410.109a	10/10/91	LD-92-017	2/12/92
410.109b	10/10/91	LD-92-017	2/12/92
410.110a	10/10/91	LD-92-006	1/24/92
410.110b	10/10/91	LD-92-006	1/24/92
410.110c	10/10/91	LD-92-006	1/24/92
410.110d	10/10/91	LD-92-006	1/24/92
410.110e	10/10/91	LD-92-006	1/24/92
410.111a	10/10/91	LD-92-006	1/24/92
410.111b	10/10/91	LD-92-006	1/24/92

410.111c	10/10/91	LD-92-006	1/24/92
410.111d	10/10/91	LD-92-006	1/24/92
410.111e	10/10/91	LD-92-006	1/24/92
410.111f	10/10/91	LD-92-006	1/24/92
410.111g	10/10/91	LD-92-006	1/24/92
410.111h	10/10/91	LD-92-006	1/24/92
410.111i	10/10/91	LD-92-006	1/24/92
410.112a	10/10/91	LD-92-006	1/24/92
410.112b	10/10/91	LD-92-006	1/24/92
410.113a	10/10/91	LD-92-006	1/24/92
410.113b	10/10/91	LD-92-006	1/24/92
410.114a	10/10/91	LD-92-006	1/24/92
410.114b	10/10/91	LD-92-006	1/24/92
410.114c	10/10/91	LD-92-006	1/24/92
410.114d	10/10/91	LD-92-006	1/24/92
410.114e	10/10/91	LD-92-006	1/24/92
410.114f	10/10/91	LD-92-006	1/24/92
410.115a	10/10/91	LD-92-006	1/24/92
410.115b	10/10/91	LD-92-006	1/24/92
410.115c	10/10/91	LD-92-006	1/24/92
410.115d	10/10/91	LD-92-006	1/24/92
410.116a	10/10/91	LD-92-006	1/24/92
410.116b	10/10/91	LD-92-006	1/24/92
410.116c	10/10/91	LD-92-006	1/24/92
410.116d	10/10/91	LD-92-006	1/24/92
410.116e	10/10/91	LD-92-006	1/24/92
410.116f	10/10/91	LD-92-006	1/24/92
410.116g	10/10/91	LD-92-006	1/24/92
410.116h	10/10/91	LD-92-006	1/24/92
410.116i	10/10/91	LD-92-006	1/24/92
410.116j	10/10/91	LD-92-006	1/24/92
410.116k	10/10/91	LD-92-006	1/24/92
410.117a	10/10/91	LD-92-006	1/24/92
410.117b	10/10/91	LD-92-006	1/24/92
410.117c	10/10/91	LD-92-006	1/24/92
410.117d	10/10/91	LD-92-006	1/24/92
410.117e	10/10/91	LD-92-006	1/24/92
410.117f	10/10/91	LD-92-006	1/24/92
410.118a	10/10/91	LD-92-006	1/24/92
410.118b	10/10/91	LD-92-006	1/24/92
410.118c	10/10/91	LD-92-006	1/24/92
410.118d	10/10/91	LD-92-006	1/24/92
410.119a	10/10/91	LD-92-006	1/24/92
410.119b	10/10/91	LD-92-006	1/24/92
410.119c	10/10/91	LD-92-006	1/24/92
410.119d	10/10/91	LD-92-006	1/24/92
410.119e	10/10/91	LD-92-006	1/24/92
410.119f	10/10/91	LD-92-006	1/24/92
410.120	10/10/91	LD-92-017	2/12/92
410.121a	10/10/91	LD-92-006	1/24/92
410.121b	10/10/91	LD-92-006	1/24/92
410.122a	10/10/91	LD-92-006	1/24/92
410.122b	10/10/91	LD-92-006	1/24/92
410.122c	10/10/91	LD-92-006	1/24/92
410.122d	10/10/91	LD-92-006	1/24/92

410.122e	10/10/91	LD-92-006	1/24/92
410.122f	10/10/91	LD-92-006	1/24/92
280.2	10/10/91	LD-92-006	1/24/92
280.3	10/10/91	LD-92-006	1/24/92
280.4a	10/10/91	LD-92-006	1/24/92
280.4b	10/10/91	LD-92-006	1/24/92
280.5	10/10/91	LD-92-006	1/24/92
280.6	10/10/91	LD-92-006	1/24/92
280.7	10/10/91	LD-92-006	1/24/92
280.8	10/10/91	LD-92-006	1/24/92
280.9	10/10/91	LD-92-006	1/24/92
280.10	10/10/91	LD-92-006	1/24/92
280.11	10/10/91	LD-92-006	1/24/92
280.12	10/10/91	LD-92-006	1/24/92
280.13	10/10/91	LD-92-006	1/24/92
280.14	10/10/91	LD-92-006	1/24/92
280.15	10/10/91	LD-92-006	1/24/92
280.16	10/10/91	LD-92-006	1/24/92
280.17	10/10/91	LD-92-006	1/24/92
280.18	10/10/91	LD-92-006	1/24/92
280.19	10/10/91	LD-92-006	1/24/92
280.20	10/10/91	LD-92-006	1/24/92
280.21	10/10/91	LD-92-006	1/24/92
280.22	10/10/91	LD-92-006	1/24/92
280.23	10/10/91	LD-92-006	1/24/92
280.24	10/10/91	LD-92-006	1/24/92
280.25	10/10/91	LD-92-006	1/24/92
280.26	10/10/91	LD-92-006	1/24/92
410.123.1	10/10/91	LD-92-006	1/24/92
410.123.2	10/10/91	LD-92-006	1/24/92
410.123.3	10/10/91	LD-92-006	1/24/92
410.124a	10/10/91	LD-92-006	1/24/92
410.124b	10/10/91	LD-92-006	1/24/92
410.124c	10/10/91	LD-92-006	1/24/92
410.124d	10/10/91	LD-92-006	1/24/92
410.124e	10/10/91	LD-92-006	1/24/92
410.125a	10/10/91	LD-92-006	1/24/92
410.125b	10/10/91	LD-92-006	1/24/92
410.126	10/10/91	LD-92-006	1/24/92
410.127	10/10/91	LD-92-006	1/24/92
410.128	10/10/91	LD-92-006	1/24/92
410.129	10/10/91	LD-92-006	1/24/92
410.129	10/10/91	LD-92-006	1/24/92
410.130a	10/10/91	LD-92-006	1/24/92
410.130b	10/10/91	LD-92-006	1/24/92
410.130c	10/10/91	LD-92-006	1/24/92
410.130d	10/10/91	LD-92-006	1/24/92
410.130e	10/10/91	LD-92-006	1/24/92
410.131a	10/10/91	LD-92-006	1/24/92
410.131b	10/10/91	LD-92-006	1/24/92
410.131c	10/10/91	LD-92-006	1/24/92
410.131d	10/10/91	LD-92-006	1/24/92
410.131e	10/10/91	LD-92-006	1/24/92
410.132a	10/10/91	LD-92-006	1/24/92
410.132b	10/10/91	LD-92-006	1/24/92

410.132c	10/10/91	LD-92-006	1/24/92
410.133a	10/10/91	LD-92-006	1/24/92
410.133b	10/10/91	LD-92-006	1/24/92
410.134a	10/10/91	LD-92-006	1/24/92
410.134b	10/10/91	LD-92-006	1/24/92
410.134c	10/10/91	LD-92-006	1/24/92
410.135a	10/10/91	LD-92-006	1/24/92
410.135b	10/10/91	LD-92-006	1/24/92
410.135c	10/10/91	LD-92-006	1/24/92
410.135d	10/10/91	LD-92-006	1/24/92
410.135e	10/10/91	LD-92-006	1/24/92
410.135f	10/10/91	LD-92-006	1/24/92
410.136a	10/10/91	LD-92-006	1/24/92
410.136b	10/10/91	LD-92-006	1/24/92
410.136c	10/10/91	LD-92-006	1/24/92
410.136d	10/10/91	LD-92-006	1/24/92
410.136e	10/10/91	LD-92-006	1/24/92
410.136f	10/10/91	LD-92-006	1/24/92
410.136g	10/10/91	LD-92-006	1/24/92
410.137a	10/10/91	LD-92-006	1/24/92
410.137b	10/10/91	LD-92-006	1/24/92
410.137c	10/10/91	LD-92-006	1/24/92
410.137d	10/10/91	LD-92-006	1/24/92
410.137e	10/10/91	LD-92-006	1/24/92
410.137f	10/10/91	LD-92-006	1/24/92
410.137g	10/10/91	LD-92-006	1/24/92
410.137h	10/10/91	LD-92-006	1/24/92
410.138a	10/10/91	LD-92-006	1/24/92
410.138b	10/10/91	LD-92-006	1/24/92
410.138c	10/10/91	LD-92-006	1/24/92
410.138d	10/10/91	LD-92-006	1/24/92
410.138e	10/10/91	LD-92-006	1/24/92
410.138f	10/10/91	LD-92-006	1/24/92
410.138g	10/10/91	LD-92-006	1/24/92
410.139a	10/10/91	LD-92-006	1/24/92
410.139b	10/10/91	LD-92-006	1/24/92
410.139c	10/10/91	LD-92-006	1/24/92
410.139d	10/10/91	LD-92-006	1/24/92
410.139e	10/10/91	LD-92-006	1/24/92
730.1a	10/10/91	LD-92-006	1/24/92
730.1b	10/10/91	LD-92-006	1/24/92
730.2	10/10/91	LD-92-006	1/24/92
780.3a	10/10/91	LD-92-006	1/24/92
780.3b	10/10/91	LD-92-006	1/24/92
780.4a	10/10/91	LD-92-006	1/24/92
780.4b	10/10/91	LD-92-006	1/24/92
780.5	10/10/91	LD-92-006	1/24/92
730.6	10/10/91	LD-91-071	12/24/91
730.7a	10/10/91	LD-92-006	1/24/92
730.8	10/10/91	LD-92-006	1/24/92
730.9a	10/10/91	LD-92-017	2/12/92
730.9b,c	10/10/91	LD-92-017	2/12/92
730.10	10/10/91	LD-92-006	1/24/92
730.11a	10/10/91	LD-92-024	2/18/92
730.11b	10/10/91	LD-92-024	2/18/92

720.1	10/30/91	LD-92-004	1/23/92
720.2	10/30/91	LD-92-001	1/31/92
720.3	10/30/91	LD-92-023	2/18/92
720.4a	10/30/91	LD-92-023	2/18/92
720.4b	10/30/91	LD-92-023	2/18/92
720.5	10/30/91	LD-92-004	1/23/92
720.6	10/30/91	LD-92-011	1/31/92
720.7	10/30/91	LD-92-004	1/23/92
720.8	10/30/91	LD-92-004	1/23/92
720.9	10/30/91	LD-92-011	1/31/92
720.10a	10/30/91	LD-92-023	2/18/92
720.10b	10/30/91	LD-92-023	2/18/92
720.10c	10/30/91	LD-92-023	2/18/92
720.11	10/30/91	LD-92-004	1/23/92
720.12	10/30/91	LD-92-004	1/23/92
720.13	10/30/91	LD-92-004	1/23/92
720.14	10/30/91	LD-92-004	1/23/92
720.15	10/30/91	LD-92-004	1/23/92
720.16	10/30/91	LD-92-004	1/23/92
720.17	10/30/91	LD-92-004	1/23/92
720.18	10/30/91	LD-92-004	1/23/92
720.19	10/30/91	LD-92-004	1/23/92
720.20	10/30/91	LD-92-011	1/31/92
720.21	10/30/91	LD-92-00	1/23/92
720.22	10/30/91	LD-92-00	1/23/92
720.23	10/30/91	LD-92-011	1/31/92
720.24	10/30/91	LD-92-00	1/23/92
720.25	10/30/91	LD-92-00	1/23/92
720.26	10/30/91	LD-92-023	2/18/92
720.27	10/30/91	LD-92-023	2/18/92
720.28	10/30/91	LD-92-004	1/23/92
720.29	10/30/91	LD-92-004	1/23/92
720.30	10/30/91	LD-92-011	1/31/92
720.31	10/30/91	LD-92-023	2/18/92
720.32	10/30/91	LD-92-004	1/23/92
720.33	10/30/91	LD-92-011	1/31/92
720.34	10/30/91	LD-92-004	1/23/92
720.35	10/30/91	LD-92-011	1/31/92
720.36	10/30/91	LD-92-011	1/31/92
720.37	10/30/91	LD-92-004	1/23/92
720.38	10/30/91	LD-92-011	1/31/92
720.39	10/30/91	LD-92-004	1/23/92
720.40	10/30/91	LD-92-004	1/23/92
720.41	10/30/91	LD-92-011	1/31/92
720.42	10/30/91	LD-92-004	1/23/92
720.43	10/30/91	LD-92-004	1/23/92
720.44	10/30/91	LD-92-004	1/23/92
720.45	10/30/91	LD-92-011	1/31/92
720.46	10/30/91	LD-92-011	1/31/92
720.47	10/30/91	LD-92-011	1/31/92
720.48	10/30/91	LD-92-011	1/31/92
720.49	10/30/91	LD-92-011	1/31/92
720.50	10/30/91	LD-92-023	2/18/92
720.51	10/30/91	LD-92-011	1/31/92
720.52	10/30/91	LD-92-011	1/31/92

720.53	10/30/91	LD-92-011	1/31/92
720.54	10/30/91	LD-92-011	1/31/92
720.55	10/30/91	LD-92-023	2/18/92
720.56	10/30/91	LD-92-011	1/31/92
720.57	10/30/91	LD-92-011	1/31/92
720.58	10/30/91	LD-92-011	1/31/92
720.59	10/30/91	LD-92-011	1/31/92
720.60	10/30/91	LD-92-011	1/31/92
720.61	10/30/91	LD-92-023	2/18/92
720.62	10/30/91	LD-92-011	1/31/92
720.63	10/30/91	LD-92-023	2/18/92
720.64	10/30/91	LD-92-011	1/31/92
720.65	10/30/91	LD-92-011	1/31/92
720.66	10/30/91	LD-92-011	1/31/92
720.67	10/30/91	LD-92-011	1/31/92
720.68	10/30/91	LD-92-011	1/31/92
720.69	10/30/91	LD-92-011	1/31/92
720.70	10/30/91	LD-92-011	1/31/92
720.71	10/30/91	LD-92-011	1/31/92
720.72	10/30/91	LD-92-011	1/31/92
720.73	10/30/91	LD-92-011	1/31/92
720.74	10/30/91	LD-92-011	1/31/92
720.75	10/30/91	LD-92-011	1/31/92
720.76	10/30/91	LD-92-011	1/31/92
720.77	10/30/91	LD-92-011	1/31/92
720.78	10/30/91	LD-92-011	1/31/92
720.79	10/30/91	LD-92-011	1/31/92
720.80	10/30/91	LD-92-011	1/31/92
720.81	10/30/91	LD-92-011	1/31/92
720.82	10/30/91	LD-92-011	1/31/92
720.83	10/30/91	LD-92-011	1/31/92
720.85	10/30/91	LD-92-023	2/18/92
720.86	10/30/91	LD-92-023	2/18/92
720.87	10/30/91	LD-92-023	2/18/92
720.88	10/30/91	LD-92-011	1/31/92
720.89	10/30/91	LD-92-011	1/31/92
720.90	10/30/91	LD-92-023	2/18/92
720.91	10/30/91	LD-92-023	2/18/92
720.92	10/30/91	LD-92-011	1/31/92
720.93	10/30/91	LD-92-011	1/31/92
720.94	10/30/91	LD-92-011	1/31/92
720.95	10/30/91	LD-92-023	2/18/92
720.96	10/30/91	LD-92-011	1/31/92
720.97	10/30/91	LD-92-011	1/31/92
720.98	10/30/91	LD-92-011	1/31/92
720.99	10/30/91	LD-92-011	1/31/92
720.100	10/30/91	LD-92-011	1/31/92
720.101	10/30/91	LD-92-011	1/31/92
721.1	10/30/91	LD-92-011	1/31/92
721.2	10/30/91	LD-92-011	1/31/92
721.3	10/30/91	LD-92-011	1/31/92
721.4	10/30/91	LD-92-011	1/31/92
721.5	10/30/91	LD-92-011	1/31/92
721.6	10/30/91	LD-92-011	1/31/92
721.7	10/30/91	LD-92-011	1/31/92

721.8	10/30/91	LD-92-011	1/31/92
721.9	10/30/91	LD-92-011	1/31/92
721.10	10/30/91	LD-92-011	1/31/92
721.11	10/30/91	LD-92-011	1/31/92
721.12	10/30/91	LD-92-011	1/31/92
721.13	10/30/91	LD-92-011	1/31/92
721.14	10/30/91	LD-92-011	1/31/92
721.15	10/30/91	LD-92-011	1/31/92
721.16	10/30/91	LD-92-011	1/31/92
721.17	10/30/91	LD-92-011	1/31/92
722.1	10/30/91	LD-92-023	2/18/92
722.2	10/30/91	LD-92-023	2/18/92
722.3	10/30/91	LD-92-023	2/18/92
722.4	10/30/91	LD-92-023	2/18/92
722.5	10/30/91	LD-92-023	2/18/92
722.6	10/30/91	LD-92-023	2/18/92
722.7	10/30/91	LD-92-023	2/18/92
722.8	10/30/91	LD-92-023	2/18/92
722.9	10/30/91	LD-92-023	2/18/92
722.10	10/30/91	LD-92-023	2/18/92
722.11	10/30/91	LD-92-023	2/18/92
722.12	10/30/91	LD-92-023	2/18/92
722.13	10/30/91	LD-92-023	2/18/92
722.14	10/30/91	LD-92-023	2/18/92
722.15	10/30/91	LD-92-023	2/18/92
722.16	10/30/91	LD-92-023	2/18/92
722.17	10/30/91	LD-92-023	2/18/92
722.18	10/30/91	LD-92-023	2/18/92
722.19	10/30/91	LD-92-023	2/18/92
722.20	10/30/91	LD-92-023	2/18/92
722.20	10/30/91	LD-92-023	2/18/92
722.21	10/30/91	LD-92-023	2/18/92
722.22	10/30/91	LD-92-023	2/18/92
722.23	10/30/91	LD-92-023	2/18/92
722.24	10/30/91	LD-92-023	2/18/92
722.25	10/30/91	LD-92-023	2/18/92
722.26	10/30/91	LD-92-023	2/18/92
722.27	10/30/91	LD-92-023	2/18/92
722.28	10/30/91	LD-92-023	2/18/92
722.29	10/30/91	LD-92-023	2/18/92
722.30	10/30/91	LD-92-023	2/18/92
722.31	10/30/91	LD-92-023	2/18/92
722.32	10/30/91	LD-92-023	2/18/92
722.33	10/30/91	LD-92-023	2/18/92
722.34	10/30/91	LD-92-023	2/18/92
722.35	10/30/91	LD-92-023	2/18/92
722.36	10/30/91	LD-92-023	2/18/92
722.37	10/30/91	LD-92-023	2/18/92
722.38	10/30/91	LD-92-023	2/18/92
722.39	10/30/91	LD-92-023	2/18/92
722.40	10/30/91	LD-92-023	2/18/92
722.41	10/30/91	LD-92-023	2/18/92
722.42	10/30/91	LD-92-023	2/18/92
722.43	10/30/91	LD-92-023	2/18/92
722.44	10/30/91	LD-92-023	2/18/92

722.45	10/30/91	LD-92-023	2/18/92
722.46	10/30/91	LD-92-023	2/18/92
722.47	10/30/91	LD-92-023	2/18/92
722.48	10/30/91	LD-92-023	2/18/92
722.49	10/30/91	LD-92-023	2/18/92
722.50	10/30/91	LD-92-023	2/18/92
722.51	10/30/91	LD-92-023	2/18/92
722.52	10/30/91	LD-92-023	2/18/92
722.53	10/30/91	LD-92-023	2/18/92
722.54	10/30/91	LD-92-023	2/18/92
722.55	10/30/91	LD-92-023	2/18/92
722.56	10/30/91	LD-92-023	2/18/92
722.57	10/30/91	LD-92-023	2/18/92
722.58	10/30/91	LD-92-023	2/18/92
722.59	10/30/91	LD-92-023	2/18/92
722.60	10/30/91	LD-92-023	2/18/92
722.61	10/30/91	LD-92-023	2/18/92
722.62	10/30/91	LD-92-023	2/18/92
722.63	10/30/91	LD-92-023	2/18/92
722.64	10/30/91	LD-92-023	2/18/92
722.65	10/30/91	LD-92-023	2/18/92
722.66	10/30/91	LD-92-023	2/18/92
722.67	10/30/91	LD-92-023	2/18/92
722.68	10/30/91	LD-92-023	2/18/92
722.69	10/30/91	LD-92-023	2/18/92
722.70	10/30/91	LD-92-023	2/18/92
722.71	10/30/91	LD-92-023	2/18/92
722.72	10/30/91	LD-92-023	2/18/92
722.73	10/30/91	LD-92-023	2/18/92
722.74	10/30/91	LD-92-023	2/18/92
722.75	10/30/91	LD-92-023	2/18/92
722.76	10/30/91	LD-92-023	2/18/92
722.77	10/30/91	LD-92-023	2/18/92
722.78	10/30/91	LD-92-023	2/18/92
722.79	10/30/91	LD-92-023	2/18/92
722.80	10/30/91	LD-92-023	2/18/92
722.81	10/30/91	LD-92-023	2/18/92
722.82	10/30/91	LD-92-023	2/18/92
722.83	10/30/91	LD-92-023	2/18/92
722.84	10/30/91	LD-92-023	2/18/92
722.85	10/30/91	LD-92-023	2/18/92
722.86	10/30/91	LD-92-023	2/18/92
722.87	10/30/91	LD-92-023	2/18/92
722.88	10/30/91	LD-92-023	2/18/92
722.89	10/30/91	LD-92-023	2/18/92
722.90	10/30/91	LD-92-023	2/18/92
722.91	10/30/91	LD-92-023	2/18/92
722.92	10/30/91	LD-92-023	2/18/92
722.93	10/30/91	LD-92-023	2/18/92
722.94	10/30/91	LD-92-008	1/29/92
		LD-92-023	2/18/92
722.95	10/30/91	LD-92-023	2/18/92
	11/21/91	LD-91-069	12/23/91
722.59	10/30/91	LD-92-023	2/18/92
722.60	10/30/91	LD-92-023	2/18/92

722.61	10/30/91	LD-92-023	2/18/92
722.62	10/30/91	LD-92-023	2/18/92
722.63	10/30/91	LD-92-023	2/18/92
722.64	10/30/91	LD-92-023	2/18/92

APPENDIX C
LIST OF ABBREVIATIONS

The following is a list of abbreviations used throughout this report and the DSER.

-A-

ac	alternating current
AAC	alternate ac
ABB	Asea Brown Boveri
ABWR	advanced boiling water reactor
ACC	advanced control complex
ACI	American Concrete Institute
ACI	auto-closure interlock
ACRS	Advisory Committee on Reactor Safeguards
ADS	atmospheric dump system
ADV	atmospheric dump valve
AFAS	alternate feedwater actuation signal
AFW	auxiliary feedwater
AFWAS	auxiliary feedwater actuation system
AHUs	air handling units
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALWR	advanced light water reactor
AMS	aerial monitoring system
AMSAC	ATWS (anticipated transient without scram) mitigating system actuation circuitry
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANSYS	General purpose finite element computer program
AOO	anticipated operational occurrence
AOVs	air-operated valves
APC	auxiliary process cabinet
APS	alternate protection system
ASB	Auxiliary Systems Branch (previous NRC organization)
ASC	aggressive secondary cooldown
ASCE	American Society of Civil Engineers
ASI	adverse system interaction
ASIS	American Society for Industrial Security
ASM	American Society for Metals
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AVS	annulus ventilation system
AWP	automatic withdrawal prohibit
AWS	American Welding Society

-B-

BAMU	boric acid make-up
BAST	boric acid storage tank
B&PV	Boiler and Pressure Vessel
BDAL	boron dilution alarm logic
BEIR	Biological Effects of Ionizing Radiation, Committee on the
BL	Bulletin
BNCS	Board on Nuclear Codes and Standards
BNL	Brookhaven National Laboratories
BOAL	boron dilution alarm
BOC	beginning of cycle
BOL	beginning-of-life
BOP	balance of plant
BTP	branch technical position
BWR	boiling-water reactor

-C-

CAP	corrective action program
CAS	central alarm system
CCDF	complementary cumulative distribution function
CCF	common-cause failure
CCFP	conditional containment failure probability
CCI	corium-concrete interaction
CCL	component control logic
CCS	component control system
CCS	condensate cleanup system
CCTV	closed-circuit television
CCVS	control complex ventilation system
CC&VS	containment cooling and ventilation system
CCW	component cooling water
CCWHXSVS	component cooling water heat exchanger structure(s) ventilation system
CCWS	component cooling water system
CCWS	condenser circulating water system
CDF	core damage frequency
CDFM	conservative deterministic failure margin
CDM	certified design material
CE	Combustion Engineering, Inc.

CEA	control element assembly
CEAC	control element assembly calculator
CEADS	control element assembly drive system
CEDM	control element drive mechanism
CEDMC	control element drive mechanism control
CEDMCS	control element drive mechanism control system
CEN	Centre d'Etudes Nucleaires (France) Centre d'Etudes de l'Energie Nucleaire (Belgium)
CEOG	Combustion Engineering Owners Group
CESSAR-DC	Combustion Engineering Standard Safety Analysis Report-Design Certification
CET	core exit thermocouple
CET	containment event tree
CETS	control element test stand
CFM	critical function monitoring
CFR	Code of Federal Regulations
CFS	cavity flooding system
CHF	critical heat flux
CHRS	containment heat removal system
CHRS	containment hydrogen recombiner system
CIAS	containment isolation actuation system
CIS	containment isolation system
CIV	containment isolation valve
CMAA	Crane Manufacturing Association of America
CMF	common-mode failure
CMI	care and maintenance instruction
CMOS	complementary metal-oxide semiconductor
CMP	configuration management plan
COL	combined license
COLSS	core operating limit supervisory system
COV	coefficient of variation
CP	construction permit
CPC(s)	core protection calculator(s)
CPG	containment performance goal
CPI	containment performance improvement
CPUs	central processing units
CR	Congressional Record
CRDM	control rod drive mechanism
CRDS	control rod drive system
CREZ	control room emergency zone
CRF	correspondence routing form
CRT(s)	cathode ray tube(s)
CSAS	containment spray actuation signal
CSB	core support barrel
CSET	containment safeguards event tree
CSNI	committee on Safety of Nuclear Installations (French)
CSS	condensate storage system
CSS	containment spray system
CSTS	condensate storage and transfer system
CSTs	condensate storage tanks
CTG	combustion turbine generator
CUF	cumulative usage factor
CVCS	chemical and volume control system
CWP	CEA withdrawal prohibit
CWS	chilled water system
CWS	circulating water system

-D-

DAC	design acceptance criteria/criterion
DBA	design-basis accident
DBE	design-basis earthquake
DBPB	design-basis pipe break
DBT	design-basis tornado
DBVS	diesel building ventilation system
dc	direct current
DCD	design control document
DCH	direct containment heating
DCM	damage control measure
DCRDR	detailed control room design review
DDOF	dynamic degrees of freedom
DEMA	Diesel Engine Manfactures Association
DE&S	Duke Engineering and Services
DESI	Duke Engineering and Services, Incorporated
DF	dilution factor
DF(s)	decontamination factor(s)
DFSS	diesel fuel storage structures
DG	diesel generator
DGBSPS	DG building sump pump system
DGEAIES	DG engine air intake and exhaust system
DGECWS	DG engine cooling water system
DGESAS	DG engine starting air systems
DGELOS	DG engine lube oil system
DGFOSTS	DG fuel oil storage and transfer system
DHR	decay heat removal
DIAS	discrete indication and alarm system
DIAS-N	DIAS for normal monitoring
DIAS-P	DIAS for postaccident monitoring
DLS	diesel loading sequencer
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOF	degrees of freedom
dp	differential pressure
DPS	data processing system
D-RAP	design-reliability assessment program
DRC	dropped rod contact
DSA	dynamic strain aging
DSDG	distribution systems design guide
DSER	draft safety evaluation report
DSIs	dedicated seal injection system
DVI	direct vessel injection
DWMS	demineralized water makeup system

-E-

EAB	exclusion area boundary
ECC	emergency core cooling
ECCS	emergency core cooling system
ECSB	emergency containment spray backup system
ECW	emergency cooling water
ECWS	essential chilled water system
EDG	emergency diesel generator
EDS	electrical distribution system
EF	error function
EFAS	emergency feedwater actuation signal
EFDS	equipment and floor drainage system
EFPD	equivalent full-power day
EFPY(s)	effective full-power year(s)
EFST	emergency feedwater storage tank
EFW	emergency feedwater
EFWP	emergency feedwater pump
EFWS	emergency feedwater system
EFWST	emergency feedwater storage tank
EHC	electrohydraulic control
EM	electromagnetic
EMI	electromagnetic interference
EOF	emergency operations facility
EOGs	emergency operations guidelines
EOL	end of life
EOPs	emergency operating procedures
EP	ethylene propylene
EPA	electrical penetration assemblies
EPGs	emergency procedure guidelines
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESD	extension shaft disconnect
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFS	engineered safety feature system
ESW	essential service water
ESWS	emergency service water system
ETSB	Effluent Treatment Systems Branch (previous NRC organization)
EUS	Eastern United States
EVSE	ex-vessel steam explosion
EW	east-west

-F-

FA	forced air
FATT	fracture appearance transition temperature
FBVS	fuel building ventilation system
FCI	fuel-coolant interaction
FMEA	failure modes and effects analysis/analyses
FO	fail open
FOA	forced oil and air
FP	fire protection
FP	fission product
FPS	fire protection system
FR	Federal Register
FRS	floor response spectra
FSAR	final safety analysis report (applicant document)
FSER	final safety evaluation report (NRC document)
FVW	fussel-vesely worth
FW	feedwater
FWCS	feedwater control system
FWLB	feedwater line break
FWPB	feedwater pipe break

-G-

GDC	general design criteria/criterion
GI	generic issue
GL	generic letter
GPM	gallon(s) per minute
GPU	General Public Utilities Corporation
GSC	gland steam condenser
GSI(s)	generic safety issue(s)
GWMS	gaseous waste management system

-H-

HCLPF	high confidence in low probability of failure
HCR	human cognitive reliability
HELB	high-energy line break
HEPA	high-efficiency particulate air
HEPs	human error probabilities
HF	human factors
HFE	human factors engineering
HFE PRM	human factors engineering program review model and acceptancecriteria
HFI	human factors interface
HFP	hot full power
HIC	high integrity containers
HJTC	heated junction thermocouple
HMS	hydrogen mitigation system
HP	high-pressure
HPME	high pressure melt ejection
HPSI	high-pressure safety injection
HRA	human reliability analysis
HSI	human systems interface
HVAC	heating, ventilation, and air conditioning
HVT	holdup volume tank

-I-

I&C(s)	instrumentation & control(s)
IAS	instrument air system
ICC	inadequate core cooling

ICCI	inadequate core cooling instrumentation
ICI	in-core instrumentation
ICN	intradivision communication network
ICSB	Instrumentation and Control Branch (previous NRC organization)
IDCOR	Industry Degraded-Core Rulemaking Program
IE	initiating event
IEEE	Institute of Electrical and Electronics Engineers
IES	Illuminating Engineering Society (IES)
IGSCC	intergranular stress-corrosion cracking
IIT	incident investigation team
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPCEA	Insulated Power Cable Engineers Association
IPF	iodine protection factor
IPSO	integrated process status overview
IRRAS	integrated reliability and risk analysis system
IR(s)	interface requirement(s)
IRWST	in-containment refueling water storage tank
ISA	integrated safety assessment
ISAP	integrated safety assessment program
ISI	inservice inspection
ISLOCA	interfacing-systems loss-of-coolant accident
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria/criterion
ITP	initial test program
ITP(s)	interface and test processor(s)
IVMS	internals vibration monitoring system
IWC	subsection of ASME Section XI Code
IWD	subsection of ASME Section XI Code
IWH	subsection of ASME Section XI Code
IWP	subsection of ASME Section XI Code
IWV	subsection of ASME Section XI Code

-K-

KAG	key assumptions and ground rules
KSB	Klein, Schanzlin and Beckner Aktiengesellschaft (Germany)
KSF	Karen Silkwood Fund
Kv	kilovolt
Kva	kilovolt-ampere
Kw	kilowatt

-L-

LBB	leak-before-break
LBHSs	large-bore hydraulic snubbers
LBLOCA	large break loss-of-coolant accident
LCL	local coincidence logic
LCLs	local control limits
LCO	limiting condition for operation
LCS	local control switch
LD	letter number (applicant's identification)
LEDs	light emitting diodes
LEL	lower electrical limit
LERs	licensee event reports
LHS	laundry and hot shower
LOCA	loss-of-coolant accident
LOCV	loss of condenser vacuum
LOFW	loss of feedwater
LOOP	loss-of-offsite power
LP	low-pressure
LPD	linear power density
LPD	local power density
LPMS	loose-parts monitoring system
LPSI	low-pressure safety injection
LPZ	low-population zone
LRFD	load and resistance factor design
LSB	large secondary side break
LSSS	limiting safety system setting
LTC	long-term cooling
LTOP	low-temperature overpressure protection
LWMS	liquid waste management system
LWR	light-water reactor

-M-

MAAP	modular accident analysis program
MAAP	material access authorization program
MACCS	MELCOR Accident Consequence Code System
MC	main condenser
MCC(s)	motor control center(s)
MCES	main condenser evacuation system
MCP	main control panel
MCRACS	main control room air conditioning system
MCR	main control room
MDC	moderator density coefficient
MDS	megawatt demand setter
MEB	Mechanical Engineering Branch (previous NRC organization)
MEV	million electron volts
MFIV	main feedwater isolation valve
MFW	main feedwater
MFWL	main feedwater line
MIS	maintenance information system
MLOCA	medium-break loss-of-coolant accident
M-MIS	man-machine interface system
MOV	motor-operated valve
MPC	maximum permissible concentration
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSL	main steam line
MSLB	main steamline break

MSSS	main steamline supply system
MSSV	main steam safety valve
MSVH	main steam valve house
MTBF	mean time between failures
MTC	moderator temperature coefficient
MTS	master transfer switch
MTSs	master transfer switches
MTTR	mean time to repair
MTU	metric ton unit
MW	megawatt
MWD	megawatt day(s)
MWT	megawatt thermal

-N-

NAVS	nuclear annex ventilation system
NCA	neutron control assembly
NCC	natural circulation cooldown
NCW	normal chilled water
NCWS	normal chilled water system
NDE	nondestructive examination
NDT	nil ductility transition
NDTT	nil ductility transition temperature
NEMA	National Electrical Manufacturers Association
NEP	non-exceedance probability
NEPIA	Nuclear Energy Property Insurance Association
NF	neutron flux
NFPA	National Fire Protection Association
NG	nitroglycerin/noble gas
NI	nuclear island
NNI	non-nuclear island
NNS	non-nuclear safety
NOP	normal operating procedure
NPF	nuclear power facility
NPOC	Nuclear Power Oversight Committee
NPRDS	nuclear plant reliability data system
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NREP	National Reliability Evaluation Program
NRR	Nuclear Reactor Regulation, Office of NRC
NS	north south
NSSFC	National Severe Storm Forecast Center
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resources Council
NUREG	NRC technical report designation
NWS	National Weather Service

-O-

OBE	operating-basis earthquake
ODCM	offsite dose calculation manual
ODFs	onsite decontamination facilities
OER	operating experience report
OER	operating experience review
OFAF	one fails, all fail
OL	operating license
OPS	onsite power system
O-RAP	operations-reliability assurance process
OSC	operations support center
OSI	operational support information
OSIP	operational support information program

-P-

PABX	private automatic business exchange
PAMI	postaccident monitoring information
PAMI	postaccident monitoring instrumentation
PASS	post-accident sampling system
PCA	primary coolant activity
PCC	primary component cooling
PCPS	pool cooling and purification system
PCS	power control system
PDS	plant damage state
PED	piping evaluation diagrams
PGA	peak ground acceleration
PGH	process gas heater
P&IDs	piping and instrumentation diagrams
PIV	pressure-indicating valve
PLC	programmable logic controller(s)
PMF	probable maximum flood
PMP	probable maximum precipitation
PNL	Pacific Northwest Laboratory
PNS	permanent non-safety
PORV	power-operated relief valve
POS	plant operational state
POV	power-operated valves
PPCS	pressurizer pressure control system
PPS	plant protection system
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PRZ	pressurizer
PSB	plant service building
PSCEA	part-strength control element assembly
PSF	performance shaping factor
PSI	pre-service inspection
PSIA	pounds per square inch absolute
PSID	preliminary safety information document
PSID	pounds per square inch differential
PSS	process sampling system
PSV	primary safety valve
PSWS	potable and sanitary water system
PTS	pressurized thermal shock
PVNGS	Palo Verde Nuclear Generating Station
PVRC	Pressure Vessel Research Council
PWR	pressurized-water reactor

-Q-

QA	quality assurance

-R-

RAI	request for additional information
RAMI	reliability, availability, maintainability, and inspectability
RAP	reliability assurance program
RAS	recirculation actuation signal
RAW	risk achievement worth
RB	reactor building
RBVS	radwaste building ventilation system
RC	reactor cavity
RC	reactor coolant
RC	release class
RCFS	reactor cavity flooding system
RCGV	reactor coolant gas vent
RCGVS	reactor coolant gas vent system
RCM	reliability-centered maintenance
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCPS	reactor coolant pump system
RCS	reactor coolant system
RDS	rapid depressurization system
RDT	reactor drain tank
RDV	rapid depressurization valve
RESAR	reference safety analysis report
RETS	radiological effluent technical specifications
RFI	request for additional information
RFM	remote field multiplexor
RG	regulatory guide
RHR	residual heat removal
RHRS	residual heat removal system
RM	radiation monitor
RM	radiation monitoring
RMS	radiation monitoring system
RMS	root-mean-square
RP	reactor power
RPC	reactor power cutback
RPCS	reactor power cutback system
RPS	reactor protective system
RPV	reactor pressure vessel
RRS	reactor recirculating system
RRW	risk reduction worth
RSB	reactor service building
RSG	rapid steam generation
RSP	remote shutdown panel
RSPT	reed switch position transmitter
RSR	remote shutdown room
RT	reactor trip
RTD	resistance temperature detector
RT_{NDT}	reference nil-ductility transition temperature
RTNSS	Regulatory Treatment of Non-Safety Systems
RTS	reactor trip system
RTSS	reactor trip switchgear system
RVLMS	reactor vessel level monitoring system
RVUH	reactor vessel upper head
RWST	refueling water storage tank

-S-

SAFDL(s)	specified acceptable fuel design limit(s)
SAMDA(s)	severe accident mitigation design alternative(s)
SAR	safety analysis report
SARA	severe accident risk assessment
SARP	Severe Accident Reduction Program
SARRP	Severe Accident Risk Reduction Program
SAS	secondary alarm system
SASA	severe accident sequence analysis/analyses
SASSI	seismic analysis for soil-structure interaction
SBCS	steam bypass control system
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SBOC	superheated blowdown outside containment
SBVS	subsphere building ventilation
SCC	stress-corrosion cracking
scfm	standard cubic feet per minute
SCIV	secondary containment isolation valve
SCL	subgroup control logic
SCM	software configuration management
SCS	shutdown cooling system
SCU	statistical combination of uncertainties
SCV	steel containment vessel
SCWS	shutdown cooling water subsystem
SDC	shield design code
SDC	shutdown cooling
SDP	software development plan
SDS	safety depressurization system
SDV	steam dump valve
SE	safety evaluation
SECY	Secretary of the Commission, Office of the NRC
SEP	Systematic Evaluation Program
SER	safety evaluation report
SERG	steam explosion review group
SERS	safety evaluation report supplement
SES	safety evaluation supplement
SFD	severe fuel damage
SFPCS	spent fuel pool cooling system
SFWS	startup feedwater system
SG	steam generator
SGA	steam generator availability

SGAS	steam generator available signal
SGB	steam generator blowdown
SGB	steam generator building
SGBS	steam generator blowdown system
SGN	Societe de Genie Nucleaire (France)
SGR	self-generating reactor
SGS	steam generator system
SGT	selective group test
SGTR	steam generator tube rupture
SGTS	standby gas treatment system
SGV	steam generator vessel
SI	safety injection
SIAS	safety injection actuation signal
SIRCP	startup of an inactive reactor coolant pump
SIS	safety injection system
SIT	safety injection tank
SJAE	steam jet-air ejector
SLB	status light box
SLB	steam line break
SLC	stress limit coefficients
SLOCA	small-break loss-of-coolant accident
SMA	seismic margins analysis
SMM	saturation margin monitor
SNL	Sandia National Laboratory
SPDS	safety parameter display system
SPLB	Plant Systems Branch
SPM	success path monitoring
SQA	software quality assurance
SR	surveillance requirements
SRM	staff requirements memorandum/memoranda
SRP	Standard Review Plan
SRS	steam relief system
SRSS	square root of the sum of the squares
SSAR	standard safety analysis report
SSC	structure, system, and component
SSE	safe shutdown earthquake
SSI	soil-structure interaction
SSSI	structure-to-soil-structure interaction
SSP	software safety plan
SSWPSVS	station service water pump structure ventilation system
SSWS	station service water system
STCP	source-term code package
STS	standard technical specifications
SV	safety valve
SV	stop valve
SWC	surge withstand capability
SWMS	solid waste management system
SWS	service water system

-T-

TBCWS	turbine building cooling water system
TBD	to be determined
TBS	turbine bypass system
TBSWS	Turbine Building service water system
TBVS	turbine bypass valve
TBVS	turbine building ventilation system
TCS	turbine control system
TG	turbine generator
TGS	turbine generator system
TGSS	turbine gland sealing system
TIs	temperature indicators
TID	total integrated dose
TLCs	trip logic calculators
TLOW	total loss of main feedwater and emergency feedwater
TMI	Three Mile Island
TORC	thermal-hydraulic analytical code
TS	technical specification
TSCACS	techncial support center air conditioning system
TSC	technical support center
TSV	turbine stop valve
TUEC	Texas Utilities Electric Company
TXX	Texas Utilities letter designation

-U-

UEL	upper electrical limit
UHJTC	unheated junction thermocouple
UHS	ultimate heat sink
UL	Underwriters Laboratories, Inc.
UO	unit operator
UO_2	uranium dioxide
URD	utility requirements document
URS	ultimate rupture strength
USI(s)	unresolved safety issue(s)
USNRC	U.S. Nuclear Regulatory Commission

-V-

VCT	volume control tank

-Z-

ZPA	zero period acceleration

APPENDIX D

ABB-CE SYSTEM 80+ DSER & FSER CONTRIBUTORS

NAME	RESPONSIBILITY
S. Ali	Structural Engineering
F. Allenspach	Quality Assurance
R. Architzel	Project Management
A. Attard	Severe Accidents
J. Bongarra	Human Factors
T. Boyce	ITAAC
H. Brammer	Mechanical Engineering
W. Burton	Plant Systems
C. Carpenter	Reliability Assurance
S. Chan	Civil/Hydrologic Engineering
T. Chandrasekaran	Plant Systems
T. Cheng	Structural Engineering
M. Chiramal	Instrumentation & Control
O. Chopra	Electrical Engineering
A. Chu	Technical Specifications
M. Clark	Secretary
T. Collins	Reactor Systems
H. Conrad	Materials Engineering
R. Correia	Reliability Assurance
S. Dembek	Project Management
D. Diec	Shutdown Risk
J. Donahew	Project Management
A. Drozd	Severe Accidents
K. Eccleston	Radiation Protection (Accidents)
R. Eckenrode	Human Factors
A. El-Bassioni	PRA
T. Essig	Radiation Protection
S. Flanders	Project Management
E. Fox	Emergency Preparedness
M. Franovich	Project Management
G. Galletti	Plant Procedures & Training
G. Georgiev	Materials Engineering
G. Giese-Koch	Geoscience
R. Gramm	Quality Assurance
S. Green	Secretary
P. Harich	Secretary
J. Guo	Plant Systems
T. Hiltz	Project Management
C. Hinson	Radiation Protection
S. Hoffman	Project Management
J. Holmes	Fire Protection
S. Hou	Mechanical Engineering
G. Hsii	Reactor Systems
Y. Huang	Mechanical Engineering
M. Hum	Materials Engineering (ISI)
W. Jensen	Plant Systems (Severe Accidents)
J. Joyce	Instrumentation & Control
G. Kelly	PRA

NAME	RESPONSIBILITY
L. Kopp	Nuclear Physics
J. Kudrick	Plant Systems (Severe Accidents)
P. Kuo	Structural, Mechanical & Materials Engineering
J. Lee	Radiation Protection
S. J. Lee	Structural Engineering
S. S. Lee	Materials & Chemical Engineering
C. Li	Plant Systems
H. Li	Instrumentation & Control
Y. Li	Mechanical Engineering
J. Lyons	Plant Systems
P. Magnanelli	Secretary
S. Magruder	Project Manager
J. Main	Technical Editing
B. Marcus	Instrumentation & Control
J. McIntyre	Mechanical Engineering
R. McIntyre	Vendor Inspection Branch
B. Mendelsohn	Safeguards
J. Monninger	Plant Systems (Severe Accidents)
P. Noonan	Licensing Assistant
D. Notley	Plant Systems (Fire Protection)
R. Palla	PRA
K. Parczewski	Chemical Engineering
L. Phillips	Reactor Systems
R. Pichumani	Geotechnical Engineering
T. Polich	Reliability Assurance
R. Ramirez	Initial Test Program
J. Ramsey	Operations Events Analysis
J. Raval	Plant Systems
F. Reinhart	Technical Specifications
R. Rothman	Geoscience
M. Rubin	Reactor Systems & Core Performance
N. Saltos	PRA
R. Sanders	Technical Editing
J. Sharkey	Reliability Assurance
J. Segala	Plant Systems
P. Shea	Licensing Assistant
K. Shembarger	Project Management
D. Smith	Human Factors
D. Smith	Materials Engineering
M. Snodderly	Plant Systems (Severe Accidents)
P. Sobel	Geoscience
J. Spraul	Quality Assurance
J. Staudenmier	Thermal Hydraulic
S. Sun	Reactor Systems
B. Sweeney	Secretary
F. Talbot	Initial Test Program
C. Tan	Structural Engineering

NAME	RESPONSIBILITY
D. Tang	Structural Engineering
D. Terao	Structural, Mechanical & Materials Engineering
D. Thatcher	Electrical Engineering
C. Thomas	Electrical Engineering
T. Tjader	Technical Specifications
H. Walker	Plant Systems
T. Wambach	Project Management
M. Waterman	Instrumentation & Control
J. Watt	Plant Systems
G. West	Human Factors
J. Wigginton	Radiation Protection
J. N. Wilson	Project Management
P. Wilson	Secretary
J. Wing	PRA
A. Young	Plant Systems (Severe Accidents)
F. Young	Safeguards
Ames Laboratory	Structural Engineering
Battelle Pacific Northwest Laboratories	Startup Testing
Brookhaven National Laboratories	Severe Accidents & Materials Engineering
Energy Technology Engineering Center	Mechanical Engineering
Lawrence Livermore National Laboratory	Geotechnical & Structural Engineering, Instrumentation and Control Systems
Sandia National Laboratories	Severe Accidents
SAIC	Plant Systems

APPENDIX E

REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

May 11, 1994

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.

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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.

- 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.

- 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

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+™ Standard Design
3. Letter dated December 31, 1993, from C.B. Brinkman, ABB-CE, to USNRC Document Control Desk, Subject: System 80+™ 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

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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

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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

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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

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 Vol. 2

2. TITLE AND SUBTITLE

Final Safety Evaluation Report Related to the Certification of the System 80+ Design, Docket No. 52-002

Chapters 15 - 22 and Appendices

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1994

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT
Safety Evaluation Report

7. PERIOD COVERED *(Inclusive Dates)*

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.)*

Associate Director for Advanced Reactors and License Renewal
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

Project Number 675, Docket Numbers 50-470 and 52-002

11. ABSTRACT *(200 words or less)*

This final safety evaluation report (FSER) documents the technical review of the System 80+ standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the System 80+ design was submitted by Combustion Engineering, Inc., now Asea Brown Boveri-Combustion Engineering (ABB-CE) as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45. System 80+ is a pressurized water reactor with a rated power of 3914 megawatts thermal (MWt) and a design power of 3992 MWt at which accidents are analyzed. Many features of the System 80+ are similar to those of ABB-CE's System 80 design from which it evolved. Unique features of the System 80+ design include: a large spherical, steel containment; an in-containment refueling water storage tank; a reactor cavity flooding system, hydrogen ignitors, and a safety depressurization system for severe accident mitigation; a combustion gas turbine for an alternate ac source; and an advanced digitally based control room. On the basis of its evaluation and independent analyses, the NRC staff concludes that ABB-CE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the System 80+ standard design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Advanced Reactors, Advanced Light Water Reactor (ALWR)
Final Safety Evaluation Report (FSER)
Evolutionary Plants
Standardization
Combined License (COL)
Final Design Approval

13. AVAILABILITY STATEMENT
Unlimited

14. SECURITY CLASSIFICATION
(This Page) Unclassified
(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

NUREG-1462, Vol. 2