ABB

April 29, 1994
LD-94-030

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: System 80+™ Information for Issue Closure

Dear Sirs:

The attachments to this letter provide revisions to CESSAR-DC and comments on the System 80+ FSER. Attachment 1 provides marked-up CESSAR-DC pages resulting from our internal consistency for the new source term. These revisions are not significant, but should be given to Mr. J. Lee. They will be formally printed in Amendment W.

Attachment 2 is a listing of references in the FSER to "Appendix A" of CESSAR-DC. Since we have converted Appendix A to Chapter 20 to be consistent with the FSER, it is recommended that the FSER be revised to reference Chapter 20 rather than Appendix A.

Attachment 3 provides proposed revisions to the FSER. Additional revisions may be warranted when we agree with NRC staff on closure of the FSER's COLA confirmatory item. Attachment 4 provides CESSAR-DC changes corresponding to the agreed-upon diesel generator allowed outage time of 14 days. These revisions should be given to Mr. M. Reinhart.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

Stanley E. Ritterbusch for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
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140017

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D032

ATTACHMENT 1

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where

γ = mole fraction of steam (s) and air (a)

M = molecular weight of steam (s) or air (a)

ρ_s = steam density

$\dot{m}_{s-spy}$ = mass condensation rate of steam due to sprays

The mathematical model described in Equations (6.5-1) through (6.5-4) is evaluated numerically to yield conservative estimates of particulate removal coefficients for the sprayed volumes.

This numerical analysis is done using the SWNAUA computer code (Reference 2). This code is based on NAUA-4 (Reference 3) which includes the following aerosol processes:

Removal:

- Gravitational settling
- Diffusional plate out

Interaction:

- Brownian coagulation
- Gravitational coagulation
- Steam condensation on particles

Transport:

- Aerosol sources
- Leakages

Note that these aerosol processes are explained further in Appendix A to Reference 1.

¶ SWNAUA is a further modification of NAUA-4 to include the effects of hygroscopicity on particle steam condensation, and removal by diffusiophoresis and sprays as additional removal processes. For conservatism, the effects of hygroscopicity have not been included in the present analysis, although the impact of having included these effects would have been significant as discussed in Section 15.6.5.5.

Qualified version doesn't include hygro. JSB

¶ One of the inputs to SWNAUA is the particle size distribution of the aerosol source. For the System 80+ Design Basis LOCA analysis these distributions are lognormal with:

- r_g = 0.075 μm, σ = 1.56 for the early release of the gap activity, and
- r_g = 0.4 μm, σ = 1.46 for the release of activity associated with fuel melt.

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These distributions are based on RAFT predictions for STEP-1 (see Reference 4).

Another of the inputs to SWNAUA are the particle densities. For the System 80+ DBA LOCA the particle densities are based on the SACHA experimental results from Reference 5. The values used are 3.7 gm/cc for the gap release and 4.6 gm/cc for the melted fuel release. In applying these densities and size distributions to the SWNAUA model for System 80+ no credit has been taken for condensation of water on the particles; this is in addition to having neglected CsOH hygroscopicity as discussed above and further below.

In order to calculate the effect of steam condensation on the spray droplets, the total amount of water removed from the containment atmosphere as a function of time in the thermal-hydraulic analysis is apportioned according to heat removal by the structures and the sprays. Only condensation on the sprays is included in the fission product aerosol removal calculation of SWNAUA; the conservatism of having made this assumption is discussed further below.

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[redacted]

Insert A

Consistent with the assumption that non-organic gaseous iodine is removed as deposited upon particulate, there has been no maximum DF applied to its removal. In fact, since organic iodine is assumed to be 0.25% of the iodine released to containment (see Table 15.6.5-2), and since

Insert B

then a residual quantity of elemental iodine airborne equal to 0.025% of the total iodine released would be negligible (i.e., <10 percent of the organic iodine contribution) in terms of impact on dose. With an assumed initial percentage of 4.75%, a DF of at least 190 would be needed to achieve the "negligible" percentage of 0.025% for elemental iodine.

With a containment free volume of 3.34 X 10^6 ft^3 and approximately 7 X 10^5 gallons of primary coolant and IRWST inventory available, the partition coefficient necessary to obtain the DF of 190 would be approximately 6800. Such a partition coefficient would be readily obtainable with a spray pH of 7, or even slightly less. In Reference ~~4~~ 6, for example, a pH of 7 limited conversion of I^- to I_2 to only 0.03% even after 24 hours exposure to 0.35 Mrad/hr of radiation (about 40% greater than that for System 80+ IRWST). To get a partition coefficient less than 6800, the conversion of I^- to I_2 would have to be greater than 0.03%; therefore, a DF of 190 is conservative for a pH of 7.

The pH of 7 is expected to be reached in approximately 2.5 hours after the LOCA (see Section 6.5.3.2). Nevertheless, it is reasonable to consider as to (1) what the impact on the dose assessment would be if the rate of pH increase were significantly slower than that yielding a pH of 7 at 2.5 hours, and (2) what the long-term impact would be of lowering pH due to acid formation. These two concerns are addressed in the following sensitivity study.

Iodine Removal Sensitivity Study

In the event that IRWST pH increases more slowly than that assumed in the DBA analysis, it is possible that the spray water could become an I_2 source once the airborne concentration falls below that corresponding to the equilibrium value for the instantaneous spray pH. To investigate this behavior, it was assumed that the TSP in the hold-up volume dissolves at a constant rate over 7.5 hours, three times the expected time period for complete dissolution. As shown on the attached Figure 6.5-5 (which shows gram-atoms of iodine airborne and the spray pH as a function of time), the elemental iodine airborne in this sensitivity study does decrease less rapidly than for the base case where depletion is assumed to be on particulate, but the organic iodine is in any case bounding.

(Conservative in the long-term based on Figure 6.2.1-16)

Regarding the formation of acids in the long-term, the acids produced by irradiation in the containment atmosphere from all sources at the end of one month converted less than 25% of the tri-sodium phosphate present initially to di-sodium phosphate. If all of the tri-sodium phosphate were to be converted, the resultant pH would be slightly greater than 6.5. At a pH of 6.5 and a temperature of ~~182~~ 191°F, the long-term airborne elemental concentration is calculated to be less than ~~10~~ 20 percent of the organic. Thus, depression of the sump pH due to the long-term production of HCl by the irradiation of the electrical insulation and the radiation-induced nitric acid formation would not impact the dose assessment. 458

In determining the effectiveness of the spray (i.e., in defining the spray "lambdas") only the effects of the spray have been considered. Diffusiophoretic deposition of aerosols on the containment structural heat sinks and sedimentation in the sprayed region have been neglected. There is considerable conservatism in having done so, particularly in the immediate post-blowdown and core quench phases in which steam condensation rates are the highest. Another important phenomenon which has neglected, as discussed above, is the hygroscopic treatment of certain fission product aerosols; e.g., CsOH. CsOH is extremely hygroscopic, and the effect of having neglected the hygroscopicity of CsOH (which makes up about 25% off the aerosol mass released to containment) is to underestimate the particulate size distribution which, in turn, leads to a low estimate of spray effectiveness.

Not all of the airborne fission products are in particulate form; iodine will also appear as I_2, HI, and organic iodides in the containment atmosphere. I_2 and HI are reactive and will tend to plate out on surfaces. The major fraction of available surface is the suspended aerosol; therefore, the same spray lambda is applied to non-organic gaseous iodine in the sprayed volume as to particulates. This is a conservative assumption because the non-organic gaseous iodine spray removal lambdas would otherwise tend to be somewhat greater.

Insert (A) ~~No~~ credit is taken for spray removal of organic iodine in ~~containment.~~ ← Insert (B)

The transient spray removal lambdas for the 10CFR100 LOCA analysis are shown on Figure 6.5-~~5~~ 6.

6.5.3.4 Available Net Positive Suction Head (NPSH)

The IRWST is the suction source for the SI pumps and CS pumps during short term injection and long term cooling modes of post-accident operation. As described in Section 6.8, the Holdup Volume Tank (HVT) performs water collection services after an accident. Spillways allow accumulated water in the HVT to spill back into the IRWST, thereby replenishing IRWST water volume during accident operations. The minimum available NPSH for the SI and CS pumps was determined based on the minimum water level in the IRWST during accident conditions. In addition, the following conservative assumptions are made:

A. Fluid conditions in the IRWST are saturated; no credit is taken for an increase in containment pressure.

REFERENCES FOR SECTION 6.5

1. "Licensing Design Basis Source Term Update for the Evolutionary Advanced Light Water Reactor", Advanced Reactor Severe Accident Program (ARSAP) Source Term Expert Group, September, 1990.

2. "SWNAUA VER02.LEV00 - Aerosol Behavior in a Condensing Atmosphere - Diffusiophoresis and Spray Version on a PC," Stone and Webster Engineering Corporation, NU-185, May, 1993 (SWEC Proprietary).

3. BUNZ, H., Koyro, Schock, W., "NAUA Mod 4: A Code for Calculating Aerosol Behavior in LWR Core Melt Accidents, Code Description and User's Manual, Preliminary Description", Laboratorium fur Aerosolphysik and Filtertechnik I, Projekt Nukleare Sicherheit, Kernforschungszentrum Karlsruhe, March, 1982.

4. 6 Beahm, E.C., C.F. Weber, and T.S. Kress, "Iodine Chemical Forms in LWR Severe Accidents," NUREG/CR-5732, ORNL/TM-11861, July 1991.

Ref 5 - Albrecht and Wild, "Review of the Main Results of the SASCHA Program on Fission Product Release under Core Melting Conditions", ANS Topical Meeting on Fission Product Behavior and Source Term Research, Snowbird, Utah, July 1984

4. 2 Im, et. al., "RAFT: A Computer Model For Formation and Transport of Fission Product Aerosols in LWR Primary Systems", ANS Topical Meeting, "Fission Product Behavior and Source Term Research", Snowbird, Utah, July 1984.

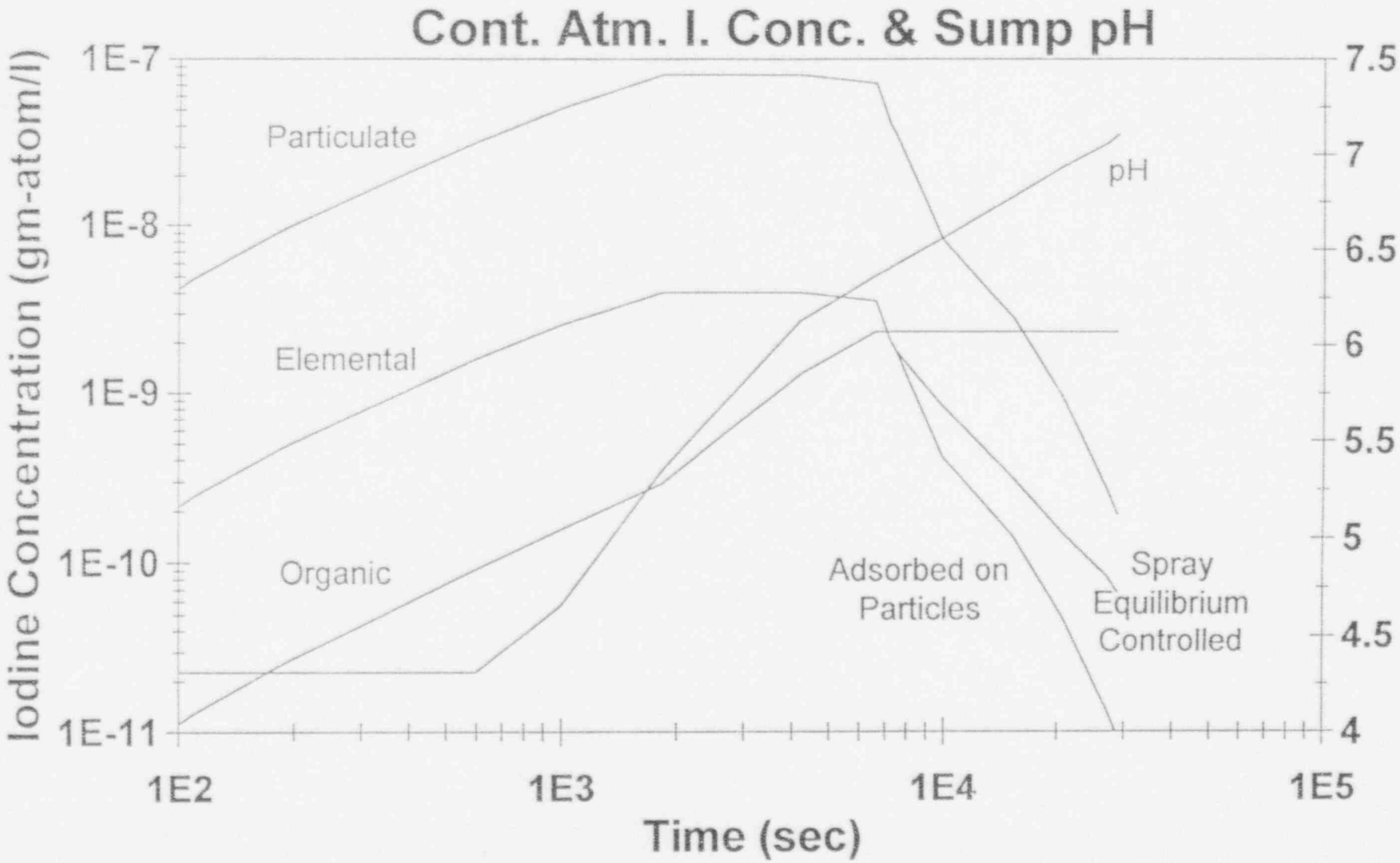

Figure 6.5-5

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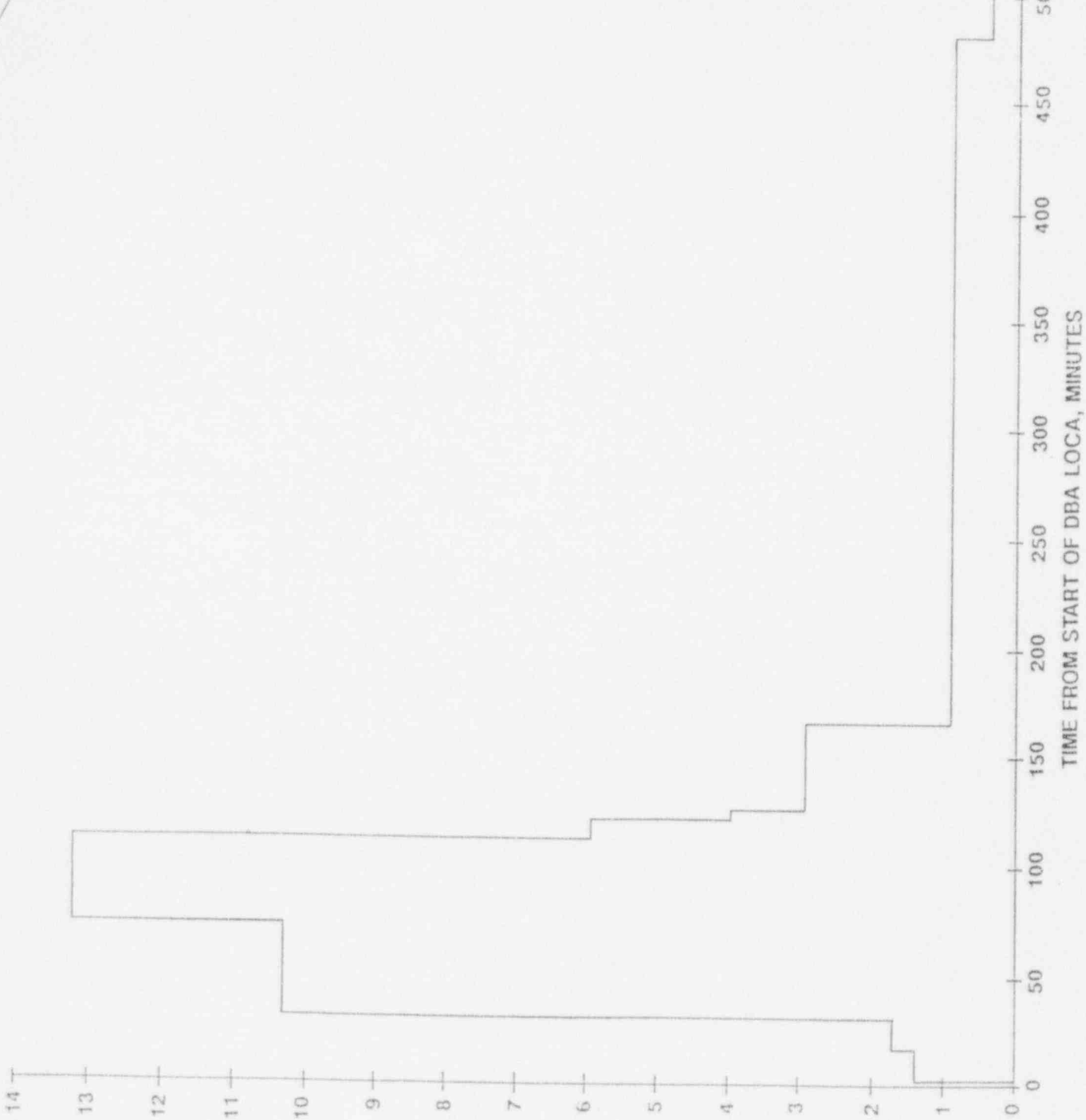

Amendment R
July 30, 1993

SYSTEM 80+	DBA LOCA SPRAY LAMBDA	Figure 6.5-5 6

ATTACHMENT 2

ADVANCED FSER "APPENDIX A" CITATIONS	
Section	**Subject** Page 1
CHRON; p 12	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 Appendix A. FICHE: 59479 293 acn: 9110300084
Ch 4; pg 22	ABB-CE described the LPMS in CESSAR-DC, Section 7.7.1.6.3 and in responses to generic safety issues B-60 and C-12 in CESSAR-DC, Appendix A.
Ch 4; pg 24	ABB-CE described the inadequate core cooling (ICC) system design in CESSAR-DC Section 7.5.1.1.7 and a response to Three Mile Island (TMI) Action Item II.F.2 in CESSAR-DC, Appendix A.
Ch 18; pg 48	In CESSAR-DC Appendix A, "Closure of Unresolved and Generic Safety Issues," ABB-CE indicates that the DPS is configured redundantly for improved reliability.
Ch 18; pg 51	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.
Ch 18; pg 53	GSI Issue HF5.1 (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.
Ch 18; pg 54	GSI Issue HF1.3.4.a (Man-Machine Interface - 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.
Ch 20; pg 2	The issues needed to meet paragraph 52.47(a)(1)(iv) are evaluated in Sections 20.1 to 20.4. Additional issues which ABB-CE considered applicable to the System 80+ design were included in Appendix A of CESSAR-DC and were evaluated by the staff.
Ch 20; pg 2	ABB-CE addressed the 50.34(f) TMI Action Plan requirements in Appendix A of CESSAR-DC. These requirements are discussed in Section 20.5 of this report.
Ch 20; pg 4	In response to DSER Open Item 20.1-1, ABB-CE provided in Amendment U to CESSAR-DC, Appendix A, general guidelines and associated references to the COL licensee for preparing plant operating and maintenance procedures to minimize the potential for water hammer.
Ch 20; pg 12	In CESSAR-DC, Appendix A and Section 3.9.3, ABB-CE also provides the general design and operability assurance acceptance criteria proposed for snubbers including large bore hydraulic snubbers.
Ch 20; pg 12	ABB-CE has addressed this issue here in the CESSAR, Appendix A and in CESSAR Section 3.9.3. The staff reviewed the proposed resolution of ABB-CE to this issue in CESSAR Appendix A and as it was applied in CESSAR Section 3.9.3; and found it acceptable.
Ch 20; pg 13	In the resolution to USI A-17 included in CESSAR-DC Appendix A, ABB-CE indicated that the System 80+ design is evaluated for its vulnerability to ASIs identified from previous designs and operating experiences reported in licensee event reports (LERs) and NRC Information Notices.

ADVANCED FSER "APPENDIX A" CITATIONS	
Section	**Subject**
Ch 20; pg 14	In Amendment I to CESSAR-DC Appendix A, ABB-CE addressed Issue A-24, stating that these methods are in accordance with the guidance of IEEE 323-1983, NUREG-0588, RG 1.89 (Rev. 1), and the generic requirements of 10 CFR 50.49 as described in CESSAR-DC Section 3.11.
Ch 20; pg 15	In the DSER, the staff stated that the approach of ABB-CE to resolving this USI [A-24] is acceptable with respect to compliance with 10 CFR 50.49, with the exception that CENPD-255-A (Rev. 3) was reviewed and approved by the staff in accordance with methods and guidance of IEEE 323-1974 and not IEEE 323-1983 as stated in Appendix A of CESSAR-DC Amendment I.
Ch 20; pg 20	In CESSAR-DC Appendix A, ABB-CE states that the System 80+ standard design is in accordance with Revision 2 of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3.
Ch 20; pg 30	In CESSAR-DC Appendix A, ABB-CE states that the methods in Appendix C of ACI 349-85 are used to treat the impactive and impulsive loads associated with a LOCA or HELB. Additionally, in response to RAI Q220.54 (Ref. 1), ABB-CE has revised the CESSAR-DC to include Positions 10 and 11 of RG 1.142 (Ref. 2) as part of the approach to resolve the issue (Ref. 3). Also, ABB-CE states in CESSAR-DC Appendix A that the containment piping analysis uses the leak-before-break (LBB) methodology to reduce the number of situations in which these loadings occur.
Ch 20; pg 30	In CESSAR-DC Appendix A, 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.
Ch 20; pg 32	In CESSAR-DC Appendix A, ABB-CE states that this is one of the issues considered to be applicable to the design of ALWR. However, after further review, ABB-CE eliminated the issue and categorized it as not relevant to the System 80+ standard design based on the staff's evaluation in NUREG-0933 which concluded that the issue was resolved with no new requirements established.

ATTACHMENT 3

1.8 Summary of Applicable Regulations and Exemptions

In accordance with Section 52.48, the staff used the applicable regulations in 10 CFR Parts 20, 50, 73, and 100 in performing its review of ABB-CE's application for design certification. During this review, the staff identified certain regulations for which application of the regulation to the System 80+ design would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. These exemptions are discussed in the sections of the SER identified below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," 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 for the advanced boiling water reactor and System 80+ standard designs. These issues included staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design. These issues were proposed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues pertaining to Evolutionary and Advanced LWR designs." The staff's positions on these issues will be identified and evaluated in the form of design-specific requirements in the staff's final SER and any supplements thereto. The completed design certification rule will then designate these positions as "applicable regulations" for the System 80+ design for the purposes of Sections 52.48, 52.54, 52.59, and 52.63. These applicable regulations are discussed in the sections of the SER identified below.

Section	Description of Exemption
3.1.1	Exemption from operating-basis earthquake design requirement.
~~[illegible]~~ 9.3.2	Exemption from post-accident sampling.

Section 2.6 "SITE PARAMETER ENVELOPE"

Tornado - System 80+ uses missile Spectrum II (alternate spectrum) of SRP Section 3.5.1.4 which deviates from the URD (Table 1.2-6, Volume II, Ref. 11) with missile Spectrum I. The approach is acceptable since both Spectra I and II meet the missile spectrum requirement of SRP Section 3.5.1.4.

Soil Properties - The minimum soil-bearing capacity of 59 ton/m^2 (12 ksf) is acceptable for the design of the ALWR. Also, sites with liquefaction potential at the site-specific SSE level are excluded. This information is reviewed and evaluated in Sections 2.5.4.8 and 2.5.4.10.

Seismology - SSE, design ground response spectra, and response time history are evaluated in Section 2.5 of this report.

In the DSER, the staff concluded that the following site parameters should be in Table 2.0-1. This was DSER Open Item 2.6-2.

Aircraft Hazards	
Plant to airport distance	8km<D<16km with annual operation less than $195D^2$ or
	D>16km with an annual operation less than $390D^2$
Plant to edge of military training routes	D≥8km with an annual operation less than 1000 flights
Plant to edge of Federal airway, holding pattern, or airport	D≥3.2km
Meteorology	1.0×10^{-3}
Short-term dilution factor	x/Q [illegible].97x10-[illegible]; EAB=500 meters
	[illegible] STET
Long-term dilution factor	x/Q [illegible].8x10-[illegible]; LPZ=3000 meters
	2.2×10^{-5}

X

X

The COL applicant referencing the System 80+ standard plant design should verify site-specific data to ensure that the data are bounded by those site envelope characteristics included in CESSAR-DC Table 2.0-1 and discussed

Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed $0.8(1.8\ S_h + S_A)$."

The System 80+ criteria are consistent with the above staff position for postulating pipe breaks and cracks. The staff concludes that the criteria meet the staff recommendations in SECY-93-087 and thus, are acceptable.

CONCRETE AND STEEL STRUCTURES

The current design practice for considering OBE and SSE ground motion effects in the seismic design of nuclear plant structures was established in the 1960s with conceptual goals of: (a) maintaining continued plant operation without damage to the structures for OBE level earthquakes and (b) ensuring safe shutdown of plant and maintaining the plant in a safe shutdown condition during and after the occurrence of an SSE. To achieve these goals, the structural responses are kept at or below the material yield stresses to preclude the on-set of plastic deformation for load combinations due to accident conditions plus the SSE. For load combinations due to operating conditions plus the OBE stresses are limited at ½ to [illegible] yield stress. The current load combinations provided in SRP Section 3.8 were developed from the above design philosophy.

Not legible

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP Sections 3.8.1, 3.8.2 and 3.8.3 load combination equations, however, shows that the major dynamic load for the overall design of structures is either the OBE or the SSE. For Category I steel and concrete structures, the staff's guidance on load combinations are provided in SRP Section 3.8.4. The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control. Similarly, an examination of the pertinent load reactor (PWR) plants and are, therefore, acceptable. Table 3.2-1, in part, identifies major components in fluid systems (such as pressure vessels, Hxs, storage tanks, pumps, piping, and valves) and in mechanical systems (such as cranes, refueling platforms, and other miscellaneous handling equipment). In addition, P&IDs in the CESSAR-DC identify the classification boundaries of interconnecting piping and valves. All of the above SSCs are constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26, previous staff positions, and applicable ASME Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety function of these systems. This constitutes the basis for satisfying GDC 1 and is, therefore, acceptable.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

All seismic Category I structures exposed to wind forces are designed to withstand the effects of the design wind specified in CESSAR-DC Table 2.0-1. Procedures used to transform the wind velocity into pressure loadings on structures are in accordance with American Society of Civil Engineers (ASCE) 7-88 (Formerly ANSI A58.1-1982), ASCE Paper 3269, and ASCE Paper 4933. The plant design with respect to capability of the structures to withstand design wind loadings is acceptable and meets the requirements of GDC 2. The design reflects, as described in SRP Section 3.3.1,

(1) appropriate consideration for the most severe wind not to exceed the velocities presented in CESSAR-DC Table 2.0-1 for future sites

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed.

These requirements are being met by the use of ASCE 7-88 (Formerly ANSI A58.1-1982) and ASCE Papers 3269 and 4933 to transform the wind velocity into tornado that exceeds the design basis tornado (DBT) should be on the order of 10^{-7} per year for each nuclear power plant. The RG delineates the maximum wind speeds of 579 kilometers per hour (km/hr) (360 miles per hour (mph)) for the Contiguous United States.

The staff reevaluated the regulatory positions in RG 1.76 using the considerable quantity of tornado data which has become available since the RG was developed. The reevaluation is discussed in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1986. The staff's interim position on RG 1.76 was issued in the March 25, 1988 letter, "ALWR Design Basis Tornado." In this interim position, the staff concluded that the maximum tornado wind speed of 531 km/hr (330 mph) is acceptable. In SECY-93-087, the staff recommended that the Commission approve its position that a maximum tornado wind speed of 483 km/hr (300 mph) is to be the design basis tornado employed in the design of evolutionary and passive ALWRs. In its SRM dated July 21, 1993, the Commission approved the staff position.

ABB-CE indicates, in the CESSAR-DC, that all seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist tornado effects in accordance with the interim staff position in RG 1.76, and the tornado missile spectrum is in accordance with SRP Section 3.5.1.4. CESSAR-DC Table 2.0-1 specifies a maximum tornado wind speed of 531 km/hr (330 mph), maximum rotational tornado wind speed of 418 km/hr (260 mph), and a maximum transnational tornado wind speed of 113 km/hr (70 mph). Also specified are a simultaneous atmospheric pressure drop to 16.5 kPa (2.4 psi) at the rate of 11.7 kPa/sec (1.7 psi/sec) and the radius of 45.7 m (150 ft). Because these parameters exceed the tornado design basis requirements specified in SECY-39-087 as approved in the July 21, 1993, SRM, the staff concludes that the ABB-CE System 80+ tornado design basis is acceptable.

The procedures used to transform the tornado wind velocity into pressure loadings are the same as for the winds discussed in Section 3.3.1 of this report. The tornado missile effects are determined using procedures discussed in CESSAR-DC Section 3.5. The tornado loadings include tornado wind pressure,

~~1992~~

3.4.1 Flood Protection

CESSAR-DC Section 3.4 states that all seismic Category I structures, components, and equipment are designed for applicable loading caused by postulated floods. Specifically, the elevation level for floods at the reactor site is determined in accordance with RG 1.59, and ANSI/ANS 2.8-~~1983~~ "Determining Design Basis Flooding at Power Reactor Sites." The finished yard grade adjacent to safety-related structures will be maintained at least 0.3 m (1 ft) below the ground floor elevation and no exterior access openings are lower than 0.3 m (1 ft) above plant grade elevation. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation. Walls subject to flooding are waterproofed and all penetrations in exterior walls up to the flood level elevation are sealed against the intrusion of water.

X

Add This Ref to Table 1.8-6

By DSER COL Action Item 3.4.1-1, the staff noted that the maximum site-specific flood levels and other safety-related structures where flood protection measures are required for the site will be addressed in the site-specific CESSAR-DC.

Subsequently, ABB-CE revised CESSAR-DC Section 2.4.1 to address this COL action item. CESSAR-DC Section 2.4.1 states, in part, that the site-specific flooding projections will consider severe precipitation, snow melt, flooding due to ice cover, river flooding, ocean flooding, tsunami flooding, seiche effects, wave and storm surge effects, hurricane effects, high lake levels, and any other effects appropriate for the specific site. CESSAR-DC Section 3.4.2 also states that the COL applicant will provide a specific description of the site and elevation for all safety-related structures, exterior accesses, equipment and systems. The staff finds that all the considerations for projecting maximum heights of site flooding events have been addressed in the CESSAR-DC. This is acceptable.

CESSAR-DC Section 3.4.4 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

By DSER COL Action Item 3.5.1.3-1, the staff noted that the COL applicant should submit a summary of the turbine maintenance and inspection program to ensure that the turbine missile generation probability will be less 10^{-4} per year. ABB-CE, in Amendment P, revised CESSAR-DC Section 3.5.1.3 to state that the COL applicant will submit a summary of the turbine maintenance and inspection program including probability calculations of turbine missile generation. This will ensure that the turbine missile generation probability be less than 10^{-4} per year for a favorably oriented turbine system (Refs. 1 and 2) and is, therefore, acceptable.

ABB-CE has sufficiently demonstrated to the staff, in accordance with RG 1.115, that the probability of turbine missile damage to SSCs important to safety is acceptably low. Therefore, the staff concludes that the turbine missile risk for the proposed plant design is acceptable and meets the requirements of GDC 4.

References

1. "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station," NUREG-1048, Supplement No. 6, July 1986.

2. Letter, from C.E. Rossi of NRC to J.A. Martin of Westinghouse, Orlando, Florida, dated February 2, 1987.

See FSER 3-32

3.5.1.4 Missiles Generated by Natural Phenomena (Tornado)

The staff reviewed the tornado-generated missiles in accordance with SRP Section 3.5.1.4. Conformance with the acceptance criteria forms the basis for the staff's evaluation of the tornado-missile spectrum with respect to the applicable regulations of 10 CFR Part 50.

The staff did not use the portions of the SRP acceptance criteria concerning missile strike probability per year to damage safety-related systems. The review for this section of the SRP is concerned with establishing the missile spectrum, not with calculating the probability of damage.

lumped masses include the weight of the floor and one-half the weight of walls directly above and below, the dead weight of known equipment and components, plus 25 percent of specified live load. Each mass point has three translational and three rotational DOF. The elastic single-stick models, with beam type elements connecting the lumped masses, were developed based on the cross-sectional properties of the structural walls between the stories they represent. The stick models also accounted for the effects of both shear and flexural deformations and the torsional effects resulting from the eccentricities between the center of mass and center of rigidity of each floor. In order to ensure that the models developed will properly simulate the dynamic behavior of the structures during an earthquake event, the frequencies calculated from the fixed-base lumped mass models were tuned by the analysis of detailed finite element models of these two buildings. In addition to the torsional DOF included in the dynamic model, an eccentricity of 5 percent of the maximum building dimension, which results in an accidental torque, is applied to the static finite element structural model to calculate the element forces due to accidental torsion. Because of the axi-symmetry of the building configurations, the soil-structural foundation systems of these two structures were represented by the two dimensional (2D) models; however, the flexibility of the foundation mats was not considered. The techniques used for modeling the seismic Category I structures (including the consideration of accidental torsion) are consistent with the guidelines of SRP Section 3.7.2 and are, therefore, acceptable. The dynamic models of these two buildings are shown in Figures 3.7C-1, 3.7C-2, and 3.7C-3 of the Appendix 3.7C to the CESSAR-DC, Amendment U.
and 3.7C-4

As discussed in CESSAR-DC Section 3.7.2 and Appendix 3.7C, Amendment U, ABB-CE performed dynamic analyses of the seismic Category I NNI structures to generate the SSE responses (structural accelerations, displacements, member forces and FRS) on a linear elastic basis. The three sets of ground motion time histories (each ground motion time history has two horizontal components and one vertical component) corresponding to the three control motions CMS1, CMS2, and CMS3 were used as input ground motions in the seismic analyses. As discussed in Section 3.7.2.1 of this report, all three components of these ground motion time histories satisfied both the response spectrum enveloping criteria and PSD enveloping criteria and are, thus, acceptable.

The seismic responses were calculated for the two horizontal directions and the vertical direction. ABB-CE did not use the technique of constant static factors for computing the vertical responses. The structural damping ratio of 7 percent for reinforced concrete structures used complies with the SSE damping value specified in RG 1.61. The techniques used for the dynamic analyses of these structures discussed above comply with the guidelines of SRP Section 3.7.2 and are, thus, acceptable. The SSI and SSSI models of these two structures are provided in CESSAR-DC Figures 3.7C-1 through 3.7C-4.

The ABB-Impell version of the SASSI computer code was used by ABB-CE to analyze the soil-structure system models and to generate the structural responses STET member forces, bending moments, for accelerations, displacements, and FRS for these two buildings. As discussed in Section 3.7.2.1 of this report, this version of the SASSI computer code has been reviewed and found acceptable by the staff. Because these two buildings are located adjacent to the NI structures and the NI structures are much heavier than the NNI structures, ABB-CE considered the effects of SSSI when the structural responses in the north-south (NS) and vertical directions of the NNI structures were calculated. The procedures applied to the NNI structures seismic analysis and design, as described in CESSAR-DC Sections 3.7.2 and 3.8.4, and Appendix 3.7C, Amendment U, are as follows:

1. In the NS direction, using the lumped-mass NNI structural model developed above and the structural model developed for the NI structures (Section 3.7.2.1 of this report), ABB-CE developed a two dimensional (2D) finite element SASSI model with the structural embedments considered and the foundation mats assumed to be rigid. This SSSI model included the structural model of the NI structures and one of the NNI structures. It also included the supporting soil foundation and adjacent soil.

2. In the east-west (EW) direction, using the lumped-mass structural model of the NNI structures, ABB-CE developed a 2D finite element dynamic model for each of the NNI structures and surrounding soil. This model did not include the SSSI effects from the NI structures.

3. Using the ABB-Impell version of SASSI computer code and the NS and vertical components of the artificial ground motion time histories (CMS1, CMS2 and CMS3), ABB-CE performed the SSSI analysis to generate structural
STET — for accelerations,
responses (member forces, bending moments, displacements and FRS) in the NS and vertical directions of the DFSS and CCW Hx structures. These seismic structural responses considered the SSSI effects.

4. Using the SASSI computer code, the finite element soil-structure system model developed in Step 2 above and the EW components of the artificial ground motion time histories (CMS1, CMS2 and CMS3), ABB-CE performed SSI analyses to generate seismic structural responses in the EW direction of these two structures. Because the DFSS and CCW Hx structures are axi-symmetrical (rectangular shape), the torsional motion about the vertical axis need not considered. To address the staff's concern regarding the use of only EW component of the three ground motion time histories as input for calculating the EW structural responses, ABB-CE demonstrated, during the January 31 through February 1, 1994 audit, that the effects of the vertical component of the three ground motions on the EW structural responses are negligible.

accelerations,
5. The NS, EW, and vertical seismic structural member forces, bending moments, and building displacement obtained from Steps 3 and 4 above were used as one of the design basis loads for the structural design and the NS, EW, and vertical FRS will be used as the input motions for the analysis of subsystems (piping systems and components) housed by these two buildings.

For the case of NNI structures founded on rock for which the SSI effects becomes negligible, a 3D fixed-based structural model was analyzed to generate structural responses in the three directions. The analysis used computer code SAP90 and the three components of the ground motion time histories corresponding to the three design response spectra, i.e., CMS1, CMS2 and CMS3, applied simultaneously. The SAP90 computer code in public domain was reviewed and validated by the staff during a previous licensing review. Therefore, the use of the SAP90 computer code for these two structures founded on rock is acceptable.

As a result of its review, the staff finds that the procedures used for calculating seismic responses, including FRS, of the DFSS and CCW Hx structures are consistent with the guidelines of SRP Section 3.7.2 and are, therefore, acceptable.

As described in CESSAR-DC Section 3.7.2 and Appendix 3.7C, Amendment U, the SSI and SSSI analyses were performed to calculate the structural responses, including FRS, of the DFSS and CCW Hx structures considered all generic site conditions use din the seismic SSI analysis of the NI structures. The structural responses (structural member forces, bending moments and displacements) corresponding to each site condition and design ground response spectrum were calculated based on the SASSI and SAP90 analyses. The final structural responses for design were calculated by enveloping all the individual responses. For the generation of the FRS envelopes, ABB-CE (1) calculated the FRS for various damping ratios at the required locations in each of the three directions, using the 2D finite element soil-structure system models or 3-D fixed base structural model developed for all site conditions and design ground motions considered (2) developed the FRS envelopes from the FRS for all site conditions an design ground motions, and (3) applied a peak broadening of ±15 percent to the FRS envelopes to account for the uncertainties associated with structural modeling, material properties, and soil dynamic moduli. Based on its review discussed in Section 3.7.2.1 of this report, the staff concludes that the use of the 12 generic site conditions for calculating the structural response envelopes of the DFSSS and CCW Hx structures covers a wide range of site conditions and provides acceptable results for the design of the DFSS and CCW Hx structures and the subsystems housed therein. As a result of its review of the above, the staff finds that ABB-CE's procedures for developing the structural response, including FRS envelopes meet the guidelines of SRP Section 3.7.2 and RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment and Components," Revision 1, and are, therefore, acceptable.

In Subsection 5.2.4 of Appendix 3.8A to the CESSAR-DC, Amendment U, and the markups of Subsection 5.1.1.3 of Appendix 3.8A to the CESSAR-DC dated February 9, 1994, ABB-CE provided the evaluation criteria and analysis procedures for the evaluation of dynamic stability (overturning, sliding and flotation) of the seismic Category I structures, including the NNI structures. In addition, ABB-CE presented its calculations for the staff review and demonstrated that the safety coefficients against building sliding, overturning and flotation are higher than 1.11. As a result of its review of the CESSAR-DC and the calculation audit conducted, the staff concludes that ABB-CE's evaluation criteria and analysis procedures are consistent with the guidelines of SRP Section 3.7.2 and are, therefore, acceptable subject to incorporating the markups discussed above into future amendments of the CESSAR-DC. This is Confirmatory Item 1.1-1.

For the evaluation of the interaction of non-safety-related structures with safety-related structures, CESSAR-DC Section 3.7.2.8, Amendment U, stated that, when the safety-related structures and non-safety-related structures are integrally connected, the non-safety-related structures are analyzed and designed as a part of the safety-related structures. If these structures are adjacent to each other, in order to ensure that the failure of a non-safety-related structure under the effect of a seismic event does not impair the integrity of the adjacent safety-related structure, the evaluation procedures are as follows:

1. sufficient separate between non-safety-related structures and safety-related structures is maintained, or

2. the non-safety-related structures are analyzed and designed to prevent their failure under SSE conditions, or

3. the safety-related structures are designed to withstand loads due to collapse of the adjacent non-safety-related structures if the separation criterion is not met.

The procedures for evaluating the interaction of non-safety-related structures with safety-related structures are consistent with the guidelines of SRP Section 3.7.2 and are acceptable.

The staff's evaluations of the seismic analyses and design of the specific NNI SSCs are discussed below.

Diesel Fuel Storage Structure (DFSS)

As described in CESSAR-DC Section 3.8.4 and Figure 1.2-16.7, Amendment U, there are two DFSSs in the System 80+ standard plant design. Each of these two buildings is a reinforced concrete box type structure with a plan dimension of 19.2m x 13.4 m (63 ft x 44 ft) and a height of 7.6 m (25 ft). The thickness of the walls, roof, and foundation mat is 0.6 m (2 ft). The embedment of the building is 4.1 m (13.5 ft) measured from the grade to the bottom of the base mat. Each of these two buildings contains two ~~by as~~ bays and each bay encloses a diesel fuel oil tank, a tank vent, a sump with a sump pump, and related piping systems.

As described in Amendment U of Appendix 3.7C t the CESSAR-DC, a dynamic model with two lumped masses connected by a massless equivalent structural member was developed to represent a DFSS in the analysis. This model coupled with the soil foundation model was used to perform the SASSI SSI and SSSI analyses or SAP90 fixed base analyses to calculate structural responses (including FRS) for each of the site conditions considered and for each set of design basis groundmotion time histories. Based on the discussion above and the audit conducted on January 31 through February 1, 1994, the staff finds that the analysis procedure and results, including the structural response envelopes and FRS envelopes, are acceptable.

Component Cooling Water Heat Exchanger Structure

Each of the two CCW Hx structures, as described in CESSAR-DC Section 3.8.4 and Figure 1.2.-16.8, Amendment U is a two-story box-type reinforced concrete building with a plan dimension of 33.5 m x 13.1m (110 ft x 43 ft) and a height of 11.6 m (38 ft). The base mat is 1.2 m (4 ft) in thickness. On the roof of the building, there are two reinforced concrete fan rooms, each located at one end of the building. The dimension of each of these two fan rooms is ~~26.2~~ 13.4 m x 7.6 m x 3 m (~~86~~ 44 ft x ~~25~~ 23 ft x 10 ft) and the thickness of the walls and roof is 0.6m (2 ft). The embedment measured from the grade to the bottom of the base mat is 5.5 m (18 ft). One of these two buildings is located at the north side and live load, thermal effects, seismic loads, and wind and tornado loads, if applicable. The design load combinations are specified in CESSAR-DC Appendix 3.9A, Section 2.3, which include normal, severe, extreme, and abnormal load combinations. The design loads and load combinations are in accordance with the guidelines of SRP 3.8.4 and therefore, acceptable. The analysis procedure and acceptance criteria for seismic Category I ductwork and supports is specified in CESSAR-DC Appendix 3.9A, Section 2.4. The damping values used are in accordance with RG 1.61 and therefore, acceptable. The effects of eccentricity of forces relative to the duct centerline is considered. The seismic analysis of seismic Category I ductwork and supports is performed using the static coefficient method, response spectrum modal analysis or the time history analysis. These methods are applied in accordance with the guidelines of SRP 3.7.3 and, therefore, are acceptable. The allowable stress criteria for seismic Category I ductwork and supports is specified in CESSAR-DC Appendix 3.9A, Section 2.5. The allowable stress criteria are established using conservative values in compliance with the requirements of American Institute of Steel Construction (AISC) standard and ANSI/AISC 690-84, as discussed in Section 3.8.4.5 of this report, and is acceptable.

The details of the cable tray/conduit and supports analysis and design procedures are provided in CESSAR-DC Appendix 3.9A, Section 3. Seismic Category I cable tray/conduit and supports are designed and supported to withstand the loads and load combinations presented in CESSAR-DC Appendix 3.9A, Section 3.2 and 3.3, respectively. The analysis and design guidelines ensure that the cable tray/conduit and supports will be within the allowable stress and deflection criteria under the design loads and load combinations. In areas where non-safety related cable tray and/or conduit passes over or near safety-related equipment or components, the tray, conduit, and support/restraint systems are design using seismic Category I criteria to prevent any damage, degration, or interference with the performance of the safety-related equipment.

The design loads for seismic Category I cable tray/conduit and supports are specified in CESSAR-DC Appendix 3.9A, Section 3.2, which include dead and live load, thermal effects, seismic loads. The design load combinations are specified in CESSAR-DC Appendix 3.9A, Section 3.3, which include normal and extreme load combinations. The design loads and load combinations are in accordance with the guidelines of SRP 3.8.4 and therefore, acceptable. The damping values are specified in CESSAR-DC Appendix 3.9A, Section 3.4 and are in accordance with RG 1.61 and, therefore, acceptable. The seismic analysis procedure for seismic Category I cable tray/conduit and supports is specified in CESSAR-DC Appendix 3.9A, Section 3.5. The seismic analysis of seismic Category I cable tray/conduit and supports is performed using the static coefficient method, response spectrum modal analysis or the time history analysis. These methods are applied in accordance with the guidelines of SRP 3.7.3 and, therefore, are acceptable. The allowable stress criteria for seismic Category I cable tray/conduit and supports is specified in CESSAR-DC Appendix 3.9A, Section 3.3 and 3.5.4. The allowable stress criteria are established using conservative values in compliance with the requirements of AISI standard for carbon steel and stainless steel cold-formed sections, and ANSI/AISC 690-84 for structural steel members, bolts and welds, as discussed in Section 3.8.4.5 of this report, and is therefore, acceptable.

The staff concludes that the design of seismic Category I ductwork and supports, cable tray and supports, and conduit and supports is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, and 4. This conclusion is based on the following:

1. ABB-CE has met the requirements of 50.55a and GDC 1 with respect to assuring that the safety related ductwork and supports and cable tray/conduit and supports are designed, fabricated, constructed, tested and inspected to quality standards commensurate with their safety function by meeting the guidelines of RGs and industry standards indicated below.

2. ABB-CE has met the requirements of 10 CFR Part 50 Appendix A, GDC 2 by designing the safety related ductwork and supports and cable tray/conduit and supports to withstand the most severe earthquake that has been established with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loading such as earthquake and other natural phenomenon.

3.8.4.2 Category I Non-Nuclear Island Structures

Four seismic Category I NNI structures fall into the scope of the CE System 80+ standard design. These four NNI structures are two DFSS and two CCW Hx structures. As described in CESSAR-DC Subsection 3.8.4.1.1, Amendment U, the DFSS and CCW Hx structures are box type reinforced concrete structures with mat foundations. One DFSS and one CCW Hx structure are located at the north side of the NI structures and the other located at the south side. The DFSS has a 19.2 m x 13.4 m (63 ft x 44 ft) plan dimension and is 7.6 m (25 ft) in height, and the plan dimension of the CCW Hx structure is 33.5 m x 13.1 m (110 ft x 43 ft) and the height is 11.6 m (38 ft). All the walls, roofs, floors and foundation mats are 0.061 m (2 ft) thick, except the CCW Hx structure foundation mat which is 1.2 m (4 ft) thick and the floor slab which is 1.0 m (3 ft) thick.

In CESSAR-DC Section 3.8.4 and Section 11.1 of Appendix 3.8A, Amendment U, ABB-CE stated tnat a static three dimensional finite element model was developed for the DFSS. Computer code ANSYS was used to analyze this structure for the combined load conditions of dead load, live loads, tornado loads (including missiles), temperature loads and SSE seismic loads. These combined load conditions were modeled as static loads in the ANSYS model. The ANSYS computer code is a public domain computer code and is has been reviewed and validated by the staff during a previous licensing review. Therefore, the use of ANSYS Code is acceptable. In addition to the design loads stated above, as described in Section 5.0, CESSAR-DC Appendix 3.8A, live loads due to precipitation (rain, snow, and ice), lateral soil pressure due to the soil density and the effects of ground water, hydrostatic loads associated with ground water and exterior flood water, and wind loads were included in the design. These loads for the DFSS design are summarized as follows:

• Maximum tornado wind speed	531.1 km/hr (330 mph)
• Tornado missiles	In accordance with SRP Section 3.5.1.4 Spectrum II, Region I

* Live loads due to precipitation 2.35 kPa (50 psf)

* Design wind speed 196.5 km/hr (122.1 mph)

When the design loads and combined load conditions were modeled, the three orthogonal components of earthquake loads (two horizontal and one vertical) were considered statically and simultaneously applied on the structures. On top of these seismic loads, an additional eccentricity of ±5 percent of the maximum building dimensions at the level under consideration was assumed to account for accidental torsion. The other design loads were also applied statically and directly on the structures.

On the basis of its review of CESSAR-DC and the audit conducted on January 31 through February 1, 1994, the staff concludes that the approach of considering the seismic loads (including dynamic soil pressure due to earthquake) and other design loads for the structural design is acceptable.

The ANSYS analysis results (structural member forces, shear forces and bending moments) form the design basis for the DFSS. A described in CESSAR-DC Subsection 3.8.4.4 and Appendix 3.8A, major materials used in the design and construction of the DFSS are concrete, reinforcing bars, and structural steel. Cement for concrete will be of Type I or II conforming to "Standard Specification for Portland Cement," ASTM C150. Aggregates for concrete will conform to "Standard Specification for Concrete Aggregate," ASTM C33. Water used in mixing concrete will be clean and free from injurious amounts of oil, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. The proposed mixing water properties will be compared with distilled water by performing the tests described in CESSAR-DC Section 3.8.4.6.1. Admixtures, if used, will conform with the applicable ASTM standard described in CESSAR-DC Section 3.8.4.6.1. In order to prevent corrosion of reinforcing bars, the combined chloride content of the admixtures and mixing water will not exceed 250 ppm. The ingredient materials will be stored in accordance with the recommendations of ACI 301 and the concrete mixes will be designed in accordance with ACI 301. Reinforcing steel will consist of deformed reinforcing bars conforming to ASTM A615, Grade 60 or ASTM A706, Grade 60. The fabrication and fabrication tolerances of reinforcing bars will be in accordance with CRSI MSP-1, "Manual of Standard Practice." The placing of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars and field tolerances will be in accordance with ACI 349-85. Epoxy coated reinforcing steel bars are used for areas where a corrosive environment is encountered. For calculating the development length, CESSAR-DC Appendix 3.8A, Section 6.2.1.1.1, states that the required splice length given in ACI 349 Section 12.2.2 shall be increased using the factors provided in ACI 318 Section 12.2.4.3. The structural steel will consist of low carbon steel conforming to ASTM A36 or other structural steels listed in ANSI/AISC N690-84. Fabrication and erection of structural steel will be in accordance with the requirements of ANSI/AISC N690-84. The welded structural connections will be in accordance with the requirements of ANSI/AISC N690-84 and bolted connections will be made with high strength bolts conforming to either ASTM A325 or A490. The quality control of materials will be in accordance with the relevant ASTM specifications and the overall QA Program described in Chapter 17 of the CESSAR-DC as supplemented by the special provisions of ACI 349-85, ASTM A615 or A706, and ANSI/AISC N690-84. The strength of the construction materials for the DFSS are as follows:

f'_c = 27.56 MPa (4000 psi) for concrete

f_y = 413.13 MPa (60000 psi) for reinforcing steel

f_y = 248.06 MPa (36000 psi) for structural steel

For the design of the CCW Hx structure, as described in the CESSAR-DC Section 3.8.4, the same design basis loads and combined load conditions and the same approach for modeling the loads to the DFSS design were considered. The construction materials and specification of the materials are the same as those used for the DFSS. Instead of modeling the structure by a 3D finite element model and using the ANSYS computer code for the analysis, formulas based on the theory of beams and plates were used and had calculations were performed for computing the structural member forces for the design. Based on

1. construction records stating material properties for concrete, reinforcing steel, and structural steel

2. as-built structural dimensions and arrangements

3. design documents for the structures

This structural analysis report will summarize the results of the reviews, evaluations and corrective actions. Deviations and design changes from the original design are acceptable provided the following acceptance criteria are met:

1. an evaluation is performed, and

2. the structural design meets the requirements specified in CESSAR-DC Section 3.8.4, and

3. the FRS of the as-built structure does not exceed the design basis FRS by more than 10 percent.

Based on the discussion above, the staff concludes that the procedures for the reconciliation analysis will ensure that the as-built DFSS and CCW Hx structures are able to withstand the structural design basis loads and combined load conditions defined in CESSAR-DC Section 3.8.4 and are, thus, acceptable.

As described in Sections 11.7 and 11.8 of Appendix 3.8A to the CESSAR-DC, Amendment U, the CCW tunnel, buried cable tunnels and conduit banks are classified as seismic Category I underground structures. The procedures and the design loads and load combinations for the analysis and design of these underground structures are discussed below:

(1) CCW Pipe Tunnel

The two CCW tunnels which are to be founded on competent structural backfill connect the CCW Hx structures to the nuclear annex. At each end of the tunnels, a gap of 10.16 cm (4 in.) with water tight rubber seal is designed between the tunnels and the adjacent structures. The tunnels have a cross-sectional dimension of 2.44m (8 ft) x 2.44m (8 ft) and the thickness of the walls, roof and foundation mat is 0.92m (3 ft). They will be designed and constructed of reinforced concrete and ACI 349 will be used with the material properties:

X f'_c=27.56 MPa (4000 psi) and f_y=413.43 MPa. (60,000 psi)

The design basis loads for the tunnels are dead loads, live loads, hydrostatic fluid pressure loads, soil static pressure loads, dynamic soil pressure due to earthquake, thermal loads, truck loads, and seismic loads. The load combinations for the design are specified in Appendix 3.8A to the CESSAR-DC. In the analysis, the tunnel was considered as a beam on an elastic foundation and the equivalent static analysis was performed. When the seismic loads were considered in the analysis and design, as discussed in Section 3.7.2 of this report above, the analysis of the buried tunnels considered the stain (axial and bending) and the associated stresses due to the effects of seismic wave passage and seismically induced differential movements of the ends of the tunnel. In addition, the ground-water effects were also considered in the design.

During the audit on January 31 through February 1, 1994, the staff raised a question on the tunnel joint details for reinforcing steels. Subsequently, ABB-CE provided the staff with the markups for the locations that those details will be provided in future CESSAR-DC amendments, as CESSAR-DC Figures 3.8B-10 and 3.8B-11. This is part of Confirmatory Item 3.8.4.2-1. Based on the discussion above and the design calculation audit conducted during January 31 through February 1, 1994, the staff concludes that the procedures for the analysis and design results of the buried tunnels are acceptable subject to resolution of the applicable part of this confirmatory item.

Check Fig numbers

Figure Nos uncertain suggest deletion

4. ABB-CE meets the requirements of GDC 5 by demonstrating that the structural systems and components are not shared between units.

5. ABB-CE meets the requirements of Appendix B because their QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the guidelines of RGs 1.69, 1.91, 1.94, 1.115, 1.142, and 1.143 and industry standards ACI-349 and ANSI/AISC N-690, "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

Not applicable

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

References

1. "System 80+ Standard Design - CESSAR-DC Design Certification," ABB Combustion Engineering, Windsor, Connecticut, Amendment I, December 21, 1990.

2. "Request for Additional Information on CESSAR-DC System 80+," Letter from T.V. Wambach of U.S. NRC to E.H. Kennedy of ABB Combustion Engineering dated September 26, 1991.

(5) ABB-CE has met the requirements of Appendix B because their QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of the NI structures foundation to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime are in conformance with established criteria, codes, standards, and RGs acceptable to the staff. These include meeting the positions of RGs 1.69, 1.94, 1.115 and 1.142 and industry standards ACI 349-85 and ANSI/AISC N690-84.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials and quality control programs; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the structures, the NI structures foundation will withstand the specified conditions without impairment of structural integrity or of the performance of required safety functions.

3.8.5.2 Category I Non-Nuclear Island Foundations

In the System 80+ design, ABB-CE employs separate reinforced-concrete mat foundations for seismic Category I NNI structures such as the DFSS and CCW Hx structures. The plan dimensions of the foundation mats for the DFSS and CCW Hx structure, as described in CESSAR-DC Section 3.8.5, are 19.2m x 13.4 m (63 ft x 44 ft) and 33.5 m x 13.1 m (110 ft x 43 ft), and the minimum thickness of these two foundation mats are 0.6 m (2 ft) and 1.2 m (4 ft), respectively.

2-3 in.

As described in CESSAR-DC Section 3.8.5, Amendment U, the reinforced concrete foundation mats of the DFSS and CCW Hx structures were analyzed and designed for the reactions due to static, seismic and all other design basis loads at the base of the superstructures, of the DFSS was modeled as a three dimensional (3D) finite element model and analyzed by computer code ANSYS. The foundation mat of the CCW Hx structure was analyzed by hand calculation. Both analyses considered the envelopes of the seismic loads calculated for all 12 site conditions and three control motions and the results obtained from the analyses together with the other design loads were used for the foundation design. The analyses and design also considered the effects of varying soil properties beneath a specific foundation mat and the effects of construction sequence, with particular emphasis on differential settlement of the foundation. To monitor the settlements of the foundation after the completion of construction, settlement monitoring devices will be installed. For the foundation design, the ACI-349 code was used. The acceptance of the ACI-349 code is discussed in Section 3.8.4.1 of this report.

As described in CESSAR-DC Sections 3.8.4.4 and 3.8.5 and Appendix 3.8A, Amendment U, major materials used in the design and construction of the DFSS are concrete, reinforcing bars, and structural steel. Cement for concrete will be of Type I or II conforming to "Standard Specifications for Portland Cement," ASTM-C150. Aggregates for concrete will conform to "Standard Specification for Concrete Aggregate," ASTM-C33. Water used in mixing concrete will be clean and free from injurious amounts of oil, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. The proposed mixing water properties will be compared with distilled water by performing the tests described in CESSAR-DC Section 3.8.4.6.1.1. Admixtures, if used, will conform with the applicable ASTM standard described in CESSAR-DC Section 3.8.4.6.1.1. In order to prevent corrosion of reinforcing bars, the combined chloride content of the admixtures and mixing water will not exceed 250 pm. The ingredient materials will be stored in accordance with the recommendations of ACI-304 and the concrete mixes will be designed in accordance with ACI-301. Reinforcing steel will consist of deformed reinforcing bars conforming to ASTM-A615, Grade 60 or ASTM-A706, Grade 60. The fabrication and fabrication tolerances of reinforcing bars will be in accordance with CRSI MSP-1, "Manual of Standard Practice." The placing of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars and field tolerances will be in accordance with ACI-349-85. Epoxy coated reinforcing steel bars are used for areas where a corrosive environment is encountered. For calculating the development length, CESSAR-DC Appendix 3.8A, Section 6.2.1.1.1, states that the required splice length given in ACI-349 Section 12.2.2 shall be increased using the factors provided in ACI-318 Section 12.2.4.3.

The quality control of materials will be in accordance with the relevant ASTM Specifications and the overall QA program described in Chapter 17 of the CESSAR-DC as supplemented by the special provisions of ACI-349-85. The strength of the construction materials for the foundations are as follows:

f'_c = 27.56 MPa (4000 psi) for concrete

f_y = 413.13 MPa (60000 psi) for reinforcing steel

From the discussion above and the design calculation audit performed on January 31 through February 1, 1994, the staff concludes that the foundation design these two buildings are acceptable. In addition to satisfying the requirements for the design loads and combined load conditions, an evaluation were performed to check the dynamic stability (sliding, overturning and floatation) of the foundations against the seismic loads. During the design calculation audit conducted on January 31 through February 1, 1994, the staff found that the safety coefficients against dynamic stability for the DFSS and CCW Hx structures are higher than 1.11 as specified in SRP Section 3.8.5. This is acceptable.

On the basis of the above review, the staff concludes that the design of the DFSS and CCW Hx structure foundations are acceptable and meets the relevant requirements of 10 CFR Part 50, and GDC 1, 2, 4, and 5. This conclusion is based on the following:

1. ABB-CE meets the requirements of GDC 1 with respect to assuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards indicated below.

Not an exemption; 50.49 calls Rev 2 "guidance".

DBA

CESSAR-DC Appendix 3.11B lists the equipment required to mitigate a DAC or to attain a safe shutdown. The CESSAR-DC states that specific equipment for each system is discussed in the appropriate section of the CESSAR-DC as reference by Appendix 3.11B. The CESSAR-DC also states that the design of the information systems important to safety will be in conformance with the guidelines of Revision 3 of RG 1.97. However, the footnote for § 50.49(b)(3) references Revision 2 for selection of the types of post-accident monitoring equipment. In issuing Revision 3, the NRC staff stated that conformance with Revision 3 would not alter the implementation for § 50.49. Therefore, conformance with Revision 2 is not required because conformance with Revision 3 meets the underlying purpose of the rule. As a result, an exemption from § 50.49(b)(3) is justified by the special circumstances set forth in § 50.12 (a)(2)(ii). Based on its review of CESSAR-DC Appendix 3.11B, the staff finds ABB-CE's approach for identifying and selecting electrical equipment required to be environmentally qualified acceptable. The staff will review specific details provided by applicants referencing the System 80+ certified design to demonstrate their compliance with 10 CFR 50.49(b)(1), (b)(2), and (b)(3) with respect to identification of electrical equipment important to safety required to be environmentally qualified. The details must include a list of systems and their components that are included in the plant environmental qualification program and design features for preventing the potential adverse consequences identified in IE Information Notice 79-22, "Qualification of Control Systems."

CESSAR-DC has elected to use the new accident source term described in draft NUREG-1465. The staff's acceptance of the new accident source term for evolutionary designs, such as CESSAR-DC System 80+, is discussed in Section 15.A.1 of this SER.

The radiation qualifications for individual safety related components are developed based on:

- The radiation environment expected at the component location from equipment installation to the end of qualified life, including the time the equipment is required to remain functional post accident, and

FSER REVIEW ITEMS

FSER Section	FSER page	Description

The following are comments on individual FSER sections. In addition, markups of these sections are provided.

5.2.4	--	No comments
6.2.3	--	No comments
6.2.4	--	No comments
6.2.5	--	No comments
6.2.6	--	No comments
6.6	--	No comments

Chapter 8:

8.2.2	8-12	See attached markup of FSER page 8-12. "13.5" should be "13.8." "Fail open" should be deleted from description of transformers.
8.5	8-69	FSER states that Combustion Turbine Generator is designed to automatically start within two minutes from the onset of a LOOP event <u>and</u> power one safety related load division within two minutes (for SBO). CTG does not automatically load safety loads. In consultations with NRC staff, the attached markup was prepared to clarify CTG starting and loading requirements.

Section 9.4:

9.4.1	9-96	Discussion of MCRACS charcoal tray and screen uses "charcoal" instead of "carbon." Also, the text encircled on page 9-96 does not agree in content with CESSAR-DC Section 9.4.1.4.D, which states: "All Main Control Room Air Conditioning System (MCRACS) ductwork outside MCREZ including the filtration units is either leak tight or is of welded construction."

FSER REVIEW ITEMS

FSER Section	FSER page	Description
9.4.1	9-97	"3,2-1" should be "3.2-1." See attached markup for proposed resolution.
9.4.1	9-100	Item 1 on FSER page 9-100 should be deleted as a confirmatory item, since the resolution to this item has already been included in Amendment U (FSER effective CESSAR-DC amendment).
9.4.1	9-100	Items 2, 3, 4, and 5 on FSER page 9-100 will be addressed in Amendment V to CESSAR-DC. Item 4 on FSER page 9-100 specifies that the "main air handling unit(s)" should be designated as "Main air conditioning unit(s)" on CESSAR-DC pages 9.4-6 and 9.4-7. The use of the word "Main" is not to be found on either of these pages. However, "air handling unit(s)" appears several times, and will be modified by Amendment V to read "air conditioning units."
9.4.2	9-101	Circled word on FSER page 9-101, "ptpand" probably should be just "and."
9.4.2	9-101&102	Last sentence of FSER page 9-101 (carries over to FSER 9-102): ABB-CE has stated in CESSAR-DC Section 9.4.2.2 that the normal mode of operation does not require any filtration and bypass dampers to be open for both the filtration trains. Given the context of Section 9.4.2.2 and the system configuration, the sentence is proposed to read: "ABB-CE has stated in CESSAR-DC Section 9.4.2.2 that the normal mode of operation does not require filtration, and the bypass dampers are open for both the filtration trains."
9.4.2	9-102	In third paragraph of FSER page 9-102, replace "charcoal" with "carbon."

FSER REVIEW ITEMS

FSER Section	FSER page	Description
9.4.3	9-105	Third paragraph on FSER page 9-105, circled sentence: change "will manually close" to "will be manually closed," as indicated in the attached markup.
9.4.3	9-105	Stray mark "·" noted, as indicated in circle in fourth paragraph on FSER page 9-105.
9.4.3	9-107	On FSER page 9-107, the confirmatory item dealing with RWBVS design data should be deleted. This is as per agreement with William Russell.
9.4.4	9-107	Second sentence of first paragraph of Section 9.4.4 should read ". . . two redundant emergency diesel generators (EDGs). . . .," instead of ". . . two emergency diesel generator (EDG). . . ."
9.4.5	9-111	Misspelled word, "upidentify," noted at the top of FSER page 9-111 (see attached markup).
9.4.5	9-112	Second paragraph on FSER page 9-112 has "non-carbon bed adsorber" specified. There is no such component in the subsphere ventilation system exhaust filter trains. Probably what was intended was "non-credited carbon adsorber." The attached markup reflects that correction.
9.4.5	9-112	Final sentence of second paragraph of FSER page 9-112, change "charcoal" to "carbon."
9.4.5	9-112	Final sentence of third paragraph of FSER page 9-112, change "charcoal" to carbon."
9.4.5	9-113	Change "in-service" to "inservice," as indicated on FSER page 9-113 markup.

FSER REVIEW ITEMS

FSER Section	FSER page	Description
9.4.5	9-114	Change "in-service" to "inservice," as indicated on FSER page 9-114 markup.
9.4.5	9-114	For the first confirmatory item on FSER page 9-114, CESSAR-DC Amendment U included the following statement in Section 9.4.5.3 (CESSAR-DC page 9.4-28): "that the HEPA filters are designed to limit the offsite does within the guidelines of 10 CFR 100." This statement will be further modified by CESSAR-DC Amendment V, which will change the word "guidelines" to "requirements." The second part of this first confirmatory item on FSER page 9-114 is to revise CESSAR-DC Section 9.4.5.1 on CESSAR-DC page 9.4-24 to state "that the SBVS is designed to limit the offsite dose following a LOCA or DBA within the requirements of 10 CFR 100," and to delete reference to SRP 6.4. The staff agreed later that the reference to SRP 6.4 was proper. However, the word "requirements" will be substituted for the currently-used term "guidelines" by Amendment V. This will comply with the staff's position on this item. The content of the second confirmatory item on FSER page 9-114 will be included at the end of the first paragraph on CESSAR-DC page 9.4-28 in Amendment V to CESSAR-DC: "The ductwork from the building exit up to an including the isolation damper are qualified for the tornado differential pressure.
9.4.6	9-115	Fourth paragraph on FSER page 9-115, first sentence should read "The low-purge subsystem relieves containment pressure during startup and shutdown." Second sentence then begins with "In-containment. . . ." See attached markup.

FSER REVIEW ITEMS

FSER Section	FSER page	Description
9.4.6	9-117	Final sentence of second paragraph of FSER page 9-117, change "charcoal" to carbon." Same applies to final sentence of third paragraph. See attached markup.
9.4.6	9-118	Confirmatory item on FSER page 9-118 is already addressed in CESSAR-DC Figure 9.4-6, which describes the dampers in question as being remotely and manually closed during a tornado warning. This item should thus be deleted from the FSER.
9.4.9	9-124	Confirmatory item on FSER page 9-124 is already addressed in CESSAR-DC Figure 9.4-8, which describes the dampers in question as being remotely and manually closed during a tornado warning. This item should thus be deleted from the FSER.
9.4.9	9-124	Change "in-service" to "inservice," as indicated on FSER page 9-124 markup.
9.4.10	9-125	Confirmatory item 1 on FSER page 9-125 should be deleted from FSER, since its requirement to revise CESSAR-DC Figure 9.4-10 to include fan status was accomplished by Amendment U to CESSAR-DC.
Section 9.5.1:		
9.5.1.2.1.4	9-141	See attached markup of FSER page 9-141. Statement deleted is not applicable.
9.5.1.2.2	9-144	See attached markup of FSER page 9-144. Statement deleted is not applicable, since fire dampers are still provided between fire areas within a division.
9.5.1.3.2	9-147	See attached markup of FSER page 9-147.

FSER REVIEW ITEMS

FSER Section	FSER page	Description
9.5.1.4.4	9-154	See attached markup of FSER page 9-154.
9.5.1.5	9-162	See attached markup of FSER page 9-162
Section 9.5.4.1:		
9.5.4.1	9-179	Circled section number on FSER page 9-179 should be "9.5.4.1.2," instead of "9.5.1.4.2."
Section 9.5.4.2		No comments
Section 9.5.5		No comments
Section 9.5.6		No comments
Section 9.5.7		No comments
Section 9.5.8		No comments
Section 9.5.9		No comments
Chapter 13:		No comments

nuclear safety grade, but is designed to withstand seismic Category I and severe accident environmental conditions. Igniters will be positioned within the containment where local pockets of hydrogen may occur during a severe accident. Each HMS igniter is an ac glow plug integrated with its own step-down transformer in a watertight enclosure that meets National Electrical Manufacturers Association Type 4 specifications.

The HMS will be manually started from the control room to accommodate the hydrogen produced by a reaction of 100 percent of fuel-clad metal with the coolant water as defined in 10 CFR 50.34(f). The igniters burn hydrogen without endangering critical equipment inside containment and maintain a hydrogen concentration below 10 percent during a postulated severe accident. The staff further discusses its review of the adequacy of HMS in Section 19.2.3.3.1 on severe accidents.

6.2.6 Containment Leakage Testing

Not an exemption — App. J allows this

ABB-CE committed to containment leakage testing for the System 80+ plant in accordance with Appendix J to 10 CFR Part 50 ~~with the following exceptions~~:

(1) The COL applicant may use the mass point leak rate test method in ANSI/ANS 56.8-1987 as an alternative to Type A testing method specified in ANSI 45.4-1972.

(2) Leaks occurring during the Type A test that could affect the test results will not prevent completion of this test if: (a) the leaks are isolated for the balance of the test; (b) the leaking component had a "pre-maintenance" local leak rate test whose results, when added to those from the Type A test, satisfy the acceptance criteria; or (c) a "post maintenance" local leak rate test of the leaking component(s) is performed and the results, when added to those from the Type A test, satisfy the acceptance criteria.

The first exception is acceptable because the current version of Section III.A.3 of Appendix J to 10 CFR Part 50 includes the ANSI/ANS 56.8-1987 method (mass point method) as an acceptable alternative.

determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the control room habitability systems meet the requirements of GDC 5, and COL Action Item 6.4-1 is resolved.

During normal and postulated accident conditions, the systems will provide: (1) controlled environment for personnel comfort and equipment operability; (2) radiation protection shielding against airborne radioactivity releases outside the control building through filtration and shielding; (3) protection against toxic releases surrounding the control building; (4) protection against the effects of high-energy line ruptures in adjacent plant areas; and (5) fire protection to ensure that the control room is manned continuously. In CESSAR-DC Chapter 15, ABB-CE describes the methods to limit the radiation exposure of control room personnel for accident conditions. The staff documents its evaluation of these methods in Section 15.4. of this report. Similarly, in CESSAR-DC Section 9.5.1, ABB-CE describes fire protection methodology, which the staff evaluated as documented in Section 9.5.1 of this DSER.

The control room emergency zone (CREZ) consists of the control room, the reactor operator office, the control room supervisor office, the emergency supplies room, the integrated plant status overview room, and the document room. In CESSAR-DC Table 3.2.1, ABB-CE identifies that the vital instrumentation and equipment rooms (including battery rooms) and the MCR air handling system components (including the air handling units, with filters, fans, ductwork, water-cooling coils, and heating coils) are Safety Class 3 and seismic Category I; will meet the quality assurance requirements of Appendix B to 10 CFR Part 50; and will remain functional after a safe shutdown earthquake. Intake and exhaust structures will be protected from tornado-generated missiles, wind-generated missiles, rain, snow, or trash.

The MCR air conditioning system is a safety related system consisting of an air conditioning system and emergency filtration system. The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one component. Redundant safety related components of the system are physically separated and protected from internally generated missiles, pipe breaks, and water sprays. The facility backup power sources

(3) An amendment to the CESSAR-DC to incorporate ABB-CE's responses to RAI Q450-3 (Section 6.4) and Q410.116 (Section 9.4.1). This was identified as Confirmatory Item 6.4-3.

By CESSAR-DC amendments, ABB-CE stated that the locations of the control room air intakes and plant unit vent are shown in the general arrangement drawings, CESSAR-DC Figures 1.2-3 and 1.2-8, and 1.2-3 and 1.2-11 respectively. Therefore, the above Confirmatory Items 6.4-1 and 6.4-2 are resolved.

Subsequently, ABB-CE incorporated responses to the request for RAIs 450.3 and 410.116 concerning the Confirmatory Item 6.4-3 issues as follows:

(1) Make up air of 0.94 m^3/sec (2000 cubic ft per minute (cfm)) is provided from the least contaminated control room air intake to offset the maximum anticipated out leakage of 0.94 m^3/sec (2000 cfm) and to pressurize the control room to a minimum of 3.2 mm (1/8-in.) water gauge positive pressure with respect to the adjacent areas.

(2) The plant unit vent and diesel building exhausts are located at least 61m (200 ft) away from the nearest control room intake (as shown in CESSAR-DC Figure 3.8-5).

(3) The CREZ volume is 1906 m^3 (67,300 ft^3) and the maximum unfiltered infiltration rate into the CREZ under accident conditions is 0.005 m^3/sec (10 cfm) (as shown in CESSAR-DC Table 15A-10). CREZ consists of the control room, reactor operator office, control room supervisor office, emergency supplies room, integrated plant status overview room, and document room.

Control logic is provided such that the less contamina intake

(4) The control complex ventilation system has two motor operated isolation dampers for the outside air intakes, two pneumatic operated (bypass) dampers designed to fail closed and redundant radiation and toxic gas monitors in each division for the filtration and toxic function. Both motor operated dampers will not be simultaneously closed on the detection of radioactive materials at both air intakes. One air intake will remain open to provide (pressurization) make-up air to balance the exfiltration.

Air intakes are protected against the effects of tornado and wind-generated missiles, rain, snow, ice or trash. The divisional filtration unit consists of a post-filter, which is a HEPA, downstream of the carbon adsorber (as shown in CESSAR-DC Figure 9.4-1).

(5) The system ductwork is leak tested in accordance with ASME N509. The battery room exhausts maintain the hydrogen concentration in the battery room below two percent.

Therefore, the above Confirmatory Item 6.4-3 is resolved.

X The air intakes are located on opposite sides of the building but are not separated by 180 degrees relative to the containment. Each outside air inlet has two isolation dampers, redundant toxic gas and radiation monitors, and a smoke detector. The emergency filtration system starts automatically if high radiation is detected at an air intake, or if a SIAS is received. If high radiation is detected at both air intakes, the automatic selection logic compares the radiation levels at each air intake and closes the isolation dampers in the air intake which has the higher reading. Therefore, outside air to pressurize the control room comes through the comparatively less contaminated inlet automatically. The pressurization and recirculation modes can also be actuated manually from the control room.

During an accident, the system operates in a pressurized mode, drawing in 0.94 m^3/sec (2000 cfm) of outside air which is mixed with 1.89 m^3/sec (4000 cfm) of air recirculated from the control room prior to being filtered by the control room filter unit. The emergency zone volume is 1906 m^3 (67,300 ft^3). The entire flow rate of 2.83 m^3/sec (6000 cfm) passes through the filter unit which includes a moisture separator, prefilter, electric preheater, absolute filter (HEPA), carbon adsorber [activated carbon depth of the carbon filter is 51 mm (2 in.)], post filter (HEPA), ducts and valves, and a fan as shown in CESSAR-DC Figures 9.4-1 and 9.4-2. The charcoal tray and screen will be all welded construction to preclude the potential loss of charcoal from adsorber cells in accordance with IE Bulletin 80-03. All ducts and equipment housings outside the CREZ are of welded construction. Flanged connections will be pressure tight and periodically visually examined and

During a LOCA, the CSS draws borated water from the IRWST and sprays to the containment atmosphere from the upper region of the containment. The sprayed water will ultimately drain back to the IRWST through the HVT. The HVT is a 227,000 L (60,000 gal) capacity stainless steel lined tank next to the IRWST with drainage spillway connections to the IRWST. The sprayed water returns to the IRWST through the spillways once the HVT water level reaches the spillway. The IRWST is a continuous water source during short term injection and long term ccoling modes of post-accident operation. There is no recirculation spray mode in this design. The effectiveness of the CSS in removing iodine from the containment is addressed in Chapter 15 of this SER.

brought to

The borated spray water contains no additive for pH control during the initial stage of a LOCA. The water in the IRWST is maintained at a minimum pH of 7 for post-LOCA iodine retention. After the blowdown, coolant from the LOCA accumulates in the HVT and starts to flow into the IRWST causing the water in the IRWST to become less acidic. Post-accident pH control of the spray water in the IRWST initially was to be provided by the granular disodium phosphate base compound which is stored in stainless steel baskets in the HVT. In Amendment R, ABB-CE changed the spray additive chemical from disodium phosphate to trisodium phosphate dodecahydrate and revised CESSAR-DC Sections 6.5.1.1, 6.5.3.2, 6.5.3.3, 16.8.5, and 16A.8.5, and Figures 6.5.4, 6.5.5, and 6.8.2, to reflect the change of spray additive. The staff finds that trisodium phosphate dodecahydrate is a stronger base alkaline than disodium phosphate. The change will enhance the capability of maintaining the IRWST water at a pH above 7 and, therefore, is acceptable.

The elevation of the baskets is above the normal operating water level in the HVT and below the IRWST spillways. During a LOCA, the baskets become immersed in water and the resulting solution overflows into the IRWST. Therefore, the spray fluid will become less acidic as the trisodium phosphate dodecahydrate mixes with the boric acid.

SRP 6.5.2, Item II.1.g, states in part, that long term iodine retention may be assumed only when the equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7. The SRP states that this pH value should be achieved by the onset of the spray recirculation mode.

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In response to the staff's concern regarding the long-term pH of the spray solution after a LOCA, ABB-CE stated that the pH will be maintained above 7, as discussed above. In addition, the containment and the CSS are designed to withstand the chemical environment imposed by spraying borated water from the IRWST and any subsequent long-term induced chemical environments. Therefore, the pH of the spray solution conforms to the SRP requirement.

Each of the two independent CSS trains has its own spray header and nozzles located in the upper part of the containment. The spray nozzles (SPRACO Company Model 1713A) are a non-clogging type and can pass particles up to 8 mm (5/16 in) diameter while covering 90 percent of the containment area with a maximum drop fall height of 25m (83 ft) and an average drop residence time of 13 seconds. The design mean drop size is 530 microns and the median drop size is 230 microns. In response to a staff question, ABB-CE submitted a histogram confirming the size distribution of the spray nozzle droplets as measured under nozzle design conditions by the manufacturer. However, ABB-CE did not specify the location of the spray nozzles and, therefore, the staff could not verify that 90-percent of the containment free volume would be covered by the sprays. The staff identified this as DSER Open Item 6.5-1.

In Amendment N, ABB-CE revised the CESSAR-DC to state that the containment spray headers and nozzles have been located inside containment such that 65 percent of the containment free volume is sprayed rather than the previous value of 90 percent. The spray removal constant for elemental iodine was re-evaluated in accordance with SRP Section 6.5.2 (Rev. 2) based on the new 65-percent volume spray coverage to yield a new value of 20 lambda per hour. ABB-CE also made major changes in CESSAR-DC Section 6.5 by adding additional subsections and revised tables and figures in conformity with the reanalyses. Subsequently, ABB-CE changed the elemental iodine spray removal constant to 10-13 lambda per hour within 30-110 minutes.

The revised CESSAR-DC contains all the pertinent information regarding the modified header and nozzle arrangement, as well as the new spray removal constant for elemental iodine. The spray volume is divided into three regions, i.e., Regions I, II, and III, which represent 82-percent of the total containment free volume. The free volumes not included as part of these components of the CSS will be accessible for maintenance, inspection, testing, and manual operation even after a DBA LOCA. All CSS-related equipment not covered by ASME Code Section III will also be accessible.

To confirm the effectiveness of the CSS, ABB-CE performed CSS failure modes and effects analysis (FMEA) and presented the results in CESSAR-DC Table 6.5-3. The analysis identified 14 different component failures. In most cases, the compensating feature of a parallel redundant containment spray path or train ensure that the spray function will not be degraded by any failure. The remaining cases have no effect on the operability of the CSS.

In CESSAR-DC Appendix 15A, ABB-CE addressed the performance of the CSS in removing fission products and presented the method it used to calculate the radiological consequences of accidents. ABB-CE recalculated the CSS fission product removal capability during a DBA LOCA based on the following input to the analysis:

Fraction of net free containment volume being sprayed	= 82 percent
Transfer rate between sprayed and unsprayed regions	= 2 volumes of unsprayed region per hour per Figure
Elemental iodine spray removal constant	= 10-13 lambda/hr within 30-110 min. (see CESSAR-DC Fig. 6.5-5) reduced by a factor of 1.26 to account for mixing within the sprayed region

The staff reviewed this analysis in accordance with SRP Section 6.5.2 and its referenced standard ANSI/ANS-56.5-1979 and confirmed the use of an elemental iodine spray removal coefficient of 13 lambda per hour which is within the limit specified in SRP 6.5.2. ABB-CE does not take any credit for organic or particulate iodine removed by the spray, while other operating plants with similar systems credit the boric acid spray for removing other chemical forms of iodine. Heat removal by sprays does not become dominant until after the first 10 minutes. Therefore, containment mixing is limited to two unsprayed volumes per hour during the first 10 minutes. The staff finds that the use of minimum mixing rate during the immediate post-blowdown period is conservative.

In the CESSAR-DC, ABB-CE credits only the CSS for removing the elemental form of iodine from the containment atmosphere after a DBA LOCA. Following a DBA LOCA, any leakage from the containment atmosphere is held up and processed in the annulus and the RB subsphere by the AVS and subsphere ventilation system (SVS) respectively. The AVS and SVS are ESF atmosphere cleanup systems and are addressed in Section 6.2.3 and Section 9.4.5 of this SER.

1.83 183 26.7 2.67

In CESSAR-DC Section 15.6.5, ABB-CE presents the offsite dose results of the DBA LOCA analysis due to containment leakage and annulus ventilation discharge. The calculated 2-hour thyroid and whole-body doses for the exclusion area radius are 1.58 Sv (158 rem) and 25.7 mSv (2.57 rem), respectively. These doses are less than the limits in 10 CFR Part 100 of 3 Sv (300 rem) thyroid and 250 mSv (25 rem) whole body and are, therefore, acceptable.

In the DSER, the staff stated that ABB-CE had not estimated the mean resident time of soluble volatile and particulate fission products in the containment building atmosphere after an accident, and had not indicated how long the continuous spray will last. Acceptance Criterion II.1.a in SRP Section 6.5.2 states, in part, that the operating period of the containment spray system should not be less than 2 hours in all cases. The staff stated that ABB-CE is required to verify the CSS post-accident operation period, which should be no less than 2 hours in all cases. This was identified as DSER Open Item 6.5-3.

ABB-CE stated that the containment spray system is designed to operate throughout the duration of a DBA, as indicated in CESSAR-DC Table 3.11B-1, up
100 days to six months without interruption. Since the operating period of the spray is far more than two hours, DSER Open Item 6.5-3 is closed.

On the basis of the above evaluation, the staff concludes that ABB-CE System 80+ containment spray system, as a fission product cleanup system, is acceptable and meets the requirements of: (1) GDC 41 with respect to the iodine removal function following a postulated loss-of-coolant accident;

The staff finds acceptable the means by which ABB-CE has addressed environmental monitoring requirements for the DIAS and DPS. This resolves DSER Open Item 7.5.3-2. The HVAC design is acceptable and discussed in Chapter 9.

ABB-CE committed to qualify equipment in harsh environments in accordance with the criteria of IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"; RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"; and IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The harsh environment is defined as including temperatures from 43 °C (110 °F) to 204 °C (400 °F); a saturated and superheated mixture of steam and air, radiation TIDs up to 4×10^5 Gy (4×10^7 rad) gamma and 2 3.5 $\times 10^6$ Gy (2 3.5 $\times 10^8$ rad) beta, and 440 ppm boric acid followed by a pH of 7.0-8.5 after 4 hours using disodium trisodium phosphate dodecahydrate. ABB-CE stated that no new harsh environment equipment will be required for the System 80+ design beyond that previously qualified for the System 80 (Palo Verde) design. This qualification conforms to the requirements of GDC 2 and 4.

The information systems important to safety conform to the guidelines for instruments to access plant conditions during and after an accident, as stated in ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," as supplemented by RG 1.97.

The design includes redundancy for both the instrument channels supplying the signal and for the displays in the control room for Category 1 variables.

sized to supply, within their self-cooled rating the most conservative power requirements of its associated Class 1E buses (switchgear, load centers, and motor control centers (MCC)), the most conservative power requirements of its associated permanent non-safety bus (switchgear, load centers, and MCCs), and power requirements of at least one RCP and its support systems. Additional margins of 33-1/3 percent and 66-2/3 percent are gained by such auxiliary cooling as forced air (FA)/forced oil ~~fail open~~ (FO)/forced oil and air (FOA) to allow for future load growth. Likewise, the unit auxiliary transformers are sized to supply within their self-~~cooled~~ rating the most conservative requirements of its two 13.8-kV non-safety buses, one 4.16-kV bus and its associated load centers and MCCs, one 4.16-kV permanent non-safety bus and two 4.16-kV Class 1E buses with their associated load centers and MCCs. Additional margins of 33-1/3 percent and 66-2/3 percent are gained by such auxiliary cooling as FA/FO/FOA to allow for future load growth.

In Amendment Q, ABB-CE revised CESSAR-DC Sections 8.1.3.B.5 and 6 to include the additional information. On this basis, the staff concludes that the unit auxiliary and reserve auxiliary transformers will have sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences and (2) the core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents. Therefore, the normal and alternate offsite power transformers will satisfy the capacity and capability requirements of GDC 17 and are acceptable. On this basis, DSER Open Item 8.2.2-2 is resolved.

In CESSAR-DC Section 8.2.1.4, ABB-CE states that all systems, equipment, and components associated with the immediate and alternate offsite power circuits have the capability of being tested during plant operation. However, in the DSER, the staff asked ABB-CE to include the following information to ensure that the requirements of GDC 18 are satisfied:

evolutionary ALWRs meet the SBO rule by including an AAC power source (e.g., CTG) of diverse design capable of powering at least one complete set of normal shutdown loads and to back up the EDGs. EPRI has also included a requirement that a large-capacity, diverse AAC power source (e.g., CTG) with the capacity to power one complete set of normal safe-shutdown loads and to back up the EDG be part of the evolutionary ALWR design.

ABB-CE committed to meet the SBO requirements by providing an AAC power source. ABB-CE stated that the AAC Source for System 80+ is a non-safety-grade combustion gas turbine provided to cope with a LOOP and an SBO scenario. This standby unit will meet the requirements in 10 CFR 50.63 by being independent and diverse from the Class 1E standby EDGs. The AAC source will not normally be directly connected to the plant's main or standby offsite power sources or to the Class 1E power distribution system, thus minimizing the possibility of a common-cause failure.

The CTG is designed to automatically start and be ready to accept loads within two minutes from the onset of a LOOP event, ~~and is designed to power one safety-related load division within two minutes (for SBO),~~ so that the plant will be capable of maintaining core cooling and containment integrity. The COL applicant will also store sufficient fuel on site to support 24 hours of CTG operation at rated load. A dedicated 125-V dc battery will power the instrumentation and controls necessary to start and run the AAC source.

ABB-CE addressed periodic testing of the AAC power source and committed to require the COL applicant to establish an AAC QA program consistent with RG 1.155, Appendix A.

Therefore, a System 80+ plant will have a fully qualified CTG as an AAC power source. However, regarding core cooling for an SBO event, ABB-CE was required to confirm that,

(1) The plant will have sufficient condensate storage to remove decay heat for the duration of an SBO in accordance with RG 1.155, Section 3.3.2.

(2) The equipment and systems will be operable during an SBO event.

any of the following events: . . . 4. Sabotage." In CESSAR-DC Section 9.2.5.2, ABB-CE states that the UHS may be a cooling pond, a river, lake, ocean, or a combination of cooling pond and lake, river, or cooling tower.

In Section 5.2.7.1 of Chapter 9 of the URD, EPRI specifies that the plant layout should avoid, if possible, having portions of the protected area perimeter abutting or crossing a body of water. ABB-CE did not address this URD provision. This was identified as DSER Open Item 9.2.5-1. By CESSAR-DC Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.5-19: "Water boundaries that form part of the protected area boundary shall be avoided, if all possible." Requiring the COL applicant to avoid, if possible, designing the protected area perimeter from abutting or crossing the UHS body of water meets Section 5.2.7.1 of Chapter 9 of the URD requirements and resolves this item.

9.2.6 Condensate Storage System

The staff reviewed the design interface requirements for the condensate storage facilities in accordance with SRP Section 9.2.6, "Condensate Storage Facilities."

The CSS provides a source of deareated condensate for makeup to the condenser and is one of the condensate sources of startup feedwater for makeup to the steam generators. It also collects and stores condensate from miscellaneous system drains. A minimum of two stainless steel condensate storage tanks (CSTs) are provided, each with a minimum capacity based on the maximum condensate usage during startup (e.g., maximum steam generator blowdown level x startup duration) plus a 100-percent margin.

The CSS is not a safety-related system since the emergency feedwater system (EFWS) is designated as the safety-grade makeup water source. Accordingly, the CSS is not seismically designed. However, ABB-CE states that leakage from the CSS or failure of the CSTs, or both will not result in unacceptable environmental effects. As noted in ABB-CE's response to RAI Q410.112, the CSTs are located in the yard and are designed in accordance with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and

This does not agree with CESSAR-DC Section 9.4.1.4.D. Also "carbon" adsorbers are used instead of "charcoal" adsorbers.

System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," as identified in CESSAR-DC Tables 9.4-3A and 9.4-5. Dampers are provided up- and downstream of each ESF filtration unit and two air-operated, fail-closed dampers are provided in the emergency circulation system bypass ducts. Each of the redundant systems is powered from independent Class 1E, diesel-backed power sources, and cooling water for the AHU is supplied from the safety-related CWS. System components are accessible for periodic inspection. The non-safety related TSCACS filter unit will satisfy the guidelines of RG 1.140, "Design, Maintenance and Testing Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," in entirety as the normal ventilation system. The MCRACS charcoal tray and screen will be all welded construction to preclude the potential loss of charcoal from adsorber cells per IE Bulletin 80-03. All ducts and equipment housings outside the CREZ of CCVS are of welded construction. Flanged connections will be pressure tight and periodically visually examined and tested to maintain at positive pressure with respect to the adjacent areas, such that, any unfiltered inleakages inside CREZ are precluded. The system is designed to maintain the infiltration rate during pressurized operation of less than 0.005 m^3/sec (10 cfm). No steam piping adjacent to CREZ air intakes or inside CREZ exists and no other HVAC system ducts other than MCR air conditioning system ducts are passing through the CREZ.

During normal operation, the inlet air is continuously monitored for radiation, toxic gas, and smoke and is mixed with return air from the control room. The control room boundary pressurization system will be periodically tested (every 18 months) to verify that the make up air required to maintain a positive minimum 3.2 mm (1/8-in) water gauge pressure inside the control room boundary with respect to the adjacent areas does not exceed 10 percent of the design value. Pressure in the control room is maintained slightly positive relative to the surrounding areas and the outdoors at all times. The system design maintains the control room and other support areas between 23 °C and 26 °C (73 °F and 78 °F) and relative humidity between 20 and 60 percent, the battery room between 15.5 °C and 32.2 °C (60 °F and 90 °F), mechanical equipment room at 40 °C (104 °F) and the remaining areas at 29.4 °C (85 °F). The provisions of the minimum instrumentation and controls for the control room filtration units are listed in CESSAR-DC Table 9.4-3A. The provisions of the other instrumentation and controls monitor locally, and/or remotely: system temperatures; filter pressure drops; damper positions; chilled water flow rates; fan air flow rates and operating status;and high radioactivity and toxic gas at each outside air intake. The system description, design parameters, and flow diagram are given in CESSAR-DC Section 9.4.1, Tables 9.4-1, 9.4-3, 9.4-3A and 3,2-1 3.2-1, and Figure 9.4-2, respectively.

The balance of control complex air conditioning systems serve the safety-related and non-safety related areas. The safety-related areas include safety-related electrical rooms, vital instrument and equipment rooms, battery rooms, and the remote shut down room. These are served by individual redundant AHUs each with roughing filters, safety-related chilled water cooling coils, and fans. The non-safety related areas include: non safety-related electrical rooms; battery rooms; operations and technical support centers; computer room; shift assembly offices; radiation access control room; casualty and security room; personnel decontamination rooms; and break room. These are served by individual air conditioning units each with a roughing filter, non-safety-related chilled water cooling coils and fan. The safety-related and non-safety related battery rooms have hydrogen detection devices to monitor hydrogen concentration. The battery room exhaust fans are designed to maintain hydrogen gas concentrations below 2 percent and their outlet ducts are located near ceiling. The redundant safety-related electrical, battery, and vital instrument and equipment room air conditioning systems are safety related and have smoke exhaust fans vented on the control building roof. Safety-related systems receive cooling water from the safety-related CWS and are served by independent Class 1E, diesel-backed power sources. System components are accessible for periodic inspection.

The emergency circulation system filtration unit starts automatically if high radiation is detected at an air intake vent or a safety injection actuation signal is received. It filters the combination of the outside air and all of the return air and delivers the filtered air to the inlet of the main air conditioning unit which maintains the proper environmental conditions in the control room. If high radiation is detected at both inlet vents, the automatic selection logic compares the radiation levels at each inlet vent and

The fuel building ventilation system is a once-through design which draws outdoor air through a damper and supply-air handling unit, supplies the air to building spaces, and exhausts the air to the outdoors through an exhaust fan. A bypass circuit of the exhaust system contains a filtration unit. The inlet supply AHU consists of a prefilter, cooling coil and electric heating coil, and a fan. This portion of the system is not safety related nor is it serviced from the Class 1E power supply. The system is designed to maintain temperature between 4.4 °C and 40 °C (40 °F and 104 °F). The inlet air vent is protected against wind and tornado missiles by missile shields above and in ? front of the opening. The system description, design parameters, ptp and flow diagram are given in CESSAR-DC Section 9.4.2, Tables 9.4-1, 9.4-3, 9.4-3A, 9.4-5, and 11.3-2, and Figure 9.4-3, respectively. The system conforms to RG 1.52 for the particulate (HEPA) filtration credited during the fuel handling accident to meet 10 CFR Part 100 limits and particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration during normal operation as identified in CESSAR-DC Tables 9.4-3 and 9.4-5 and 11.3-2 respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the FBVS. The safety-related equipment, fans, dampers, coils and ductwork will be designed and tested in accordance with ASME/ANSI N509 AG-1 "Nuclear Power Plant Air-Cleaning Units and Components," N510, "Testing of Nuclear Air Treatment Systems," codes and standards. The radiological consequences resulting from gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report.

The exhaust portion of the system is safety-related (engineered safety feature system) comprising two redundant 100-percent trains of fans and filtration units. During normal operation, air is released to the atmosphere through an exhaust fan and two control dampers. ABB-CE, in response to the staff's RAI Q410.117, stated that the single-bypass damper for the filtration system will be administratively locked closed and the system will be in operation whenever irradiated fuel handling operations above or in the fuel pool are in progress. This response was not acceptable since the single-failure criteria for these components must be met to prevent inadvertent release of radioactive contaminants to the environment. This was an DSER Open Item 9.4.2-1 in the DSER. ABB-CE has stated in CESSAR-DC Section 9.4.2.2 that the normal mode of

The fuel building ventilation system is a once-through design which draws outdoor air through a damper and supply-air handling unit, supplies the air to building spaces, and exhausts the air to the outdoors through an exhaust fan. A bypass circuit of the exhaust system contains a filtration unit. The inlet supply AHU consists of a prefilter, cooling coil and electric heating coil, and a fan. This portion of the system is not safety related nor is it serviced from the Class 1E power supply. The system is designed to maintain temperature between 4.4 °C and 40 °C (40 °F and 104 °F). The inlet air vent is protected against wind and tornado missiles by missile shields above and in front of the opening. The system description, design parameters, ptpand flow diagram are given in CESSAR-DC Section 9.4.2, Tables 9.4-1, 9.4-3, 9.4-3A, 9.4-5, and 11.3-2, and Figure 9.4-3, respectively. The system conforms to RG 1.52 for the particulate (HEPA) filtration credited during the fuel handling accident to meet 10 CFR Part 100 limits and particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration during normal operation as identified in CESSAR-DC Tables 9.4-3 and 9.4-5 and 11.3-2 respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the FBVS. The safety-related equipment, fans, dampers, coils and ductwork will be designed and tested in accordance with ASME/ANSI N509 AG-1 "Nuclear Power Plant Air-Cleaning Units and Components," N510, "Testing of Nuclear Air Treatment Systems," codes and standards. The radiological consequences resulting from gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report.

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See Next page (9-102)

operation does not require any filtration, and The bypass dampers ~~to be~~ are open for both the filtration trains. Upon receipt of a high radiation signal, the system will realign the designated filtration train automatically to the filtration mode, to comply with 10 CFR Part 20 and 10 CFR Part 50, Appendix I requirements, by opening filtration unit inlet and outlet dampers and closing bypass dampers. Switchover between trains is accomplished manually. Prior to any fuel building operations, the system is manually realigned to the filtration mode and the bypass dampers are administratively locked closed. In this mode both the filtration trains are aligned to process the effluent discharge prior to releasing through the monitored plant unit vent. The FBVS has two redundant 100-percent capacity filtration trains which meets the single failure criterion and fan and motor operated dampers in each train are powered from a separate train of the emergency Class IE standby power in the event of any single active failure. The planned administrative isolation of the bypass dampers is not considered as an active function, and based upon the above, a single bypass damper in each train would continue to meet the single failure criterion design for the exhaust side of the FBVS. Therefore, DSER Open Item 9.4.2-1 is resolved.

The CESSAR-DC Tables 9.4-3, Input for Release Analysis Filter Efficiencies, shows the creditable HEPA efficiency of 99-percent for the fuel-handling accident analysis. ABB-CE stated in CESSAR-DC Section 9.4.2.1 that the dose analysis to support 10 CFR Part 100 limits following a fuel-handling accident only takes credit for the HEPA filter and no credit is taken for the charcoal adsorber.

The staff concluded in Section 15.A.11, that with respect to the radiological consequences of potential fuel-handling accidents, credit is given for the removal of particulate iodines only. Therefore, ~~charcoal~~ Carbon adsorbers need not be credited in the FBVS.

A non-safety-related radiation monitor is located in the exhaust ductwork, upstream of the filter train inlet, which automatically directs the air through a filtration unit on detection of radioactivity in the duct. There is only one radiation detector provided which is consistent with the guidance of RG 1.97 "Instrumentation for LWR Nuclear Power Plants to Assess Plant and exhaust

Environs Conditions During and Following Accident." The redundant exhaust filtration units consists of a moisture eliminator, prefilter, electric preheater, absolute (HEPA), carbon adsorber [activated carbon depth of the carbon filter is 51 mm (2 in.), post filter (HEPA), ducts and valves, and a fan as shown in CESSAR-DC Figures 9.4-1 and 9.4-3. Each division of the air exhaust portion of the system has the capability to maintain the fuel handling and fuel storage areas at a negative pressure with respect to the atmosphere.

In the event of a fire, the exhaust and supply fans can be used for smoke removal. The fire dampers with fusible links in HVAC ductwork close under air flow conditions.

As identified in CESSAR-DC Section 9.4.2.1 and Table 3.2-1, the system is located completely within seismic Category I structure, and all safety-related components (exhaust system and associated duct-work and filter train and fans) are seismic Category I, Safety Class 3, and the quality assurance requirements of 10 CFR Part 50 Appendix B are applicable. The flood protection, protection against internally and externally generated missiles, and high- and moderate-energy pipe breaks are evaluated in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report. The equipment design and testing conforms to the requirements of ANSI/ASME AG-1, N509 and N510, and the equipment is available following an LOOP. Fan operating status and air flow rate, damper position, air temperatures, filter pressure drop, and chilled water temperatures are monitored and indication is provided either locally or in the control room. Failure of non-safety-related components does not compromise function of safety-related components.

ABB-CE committed to incorporate the associated changes in CESSAR-DC Section 9.4.2 and Tables 8.3.1-2 and 8.3.1-3 provided in response to staff RAI Q410.1-17. This was Confirmatory Item 9.4.2-1 in the DSER. By CESSAR-DC amendments, ABB-CE stated that the CESSAR-DC Tables 8.3.1-2 and 8.1.3-3 are revised to list the FBVS fans and filtration trains electric heaters. A HEPA filter in each filtration train, downstream of carbon adsorber, is provided. The testing of the FBVS safety-related equipment will be in accordance with the ASME N509, N510, and AG-1 standards. An in-service program will be implemented in accordance with 10 CFR Part 50, Appendix B and ASME Section XI, the environment to ensure that all potentially radioactive releases are monitored before discharge. The system is a once-through cycle type. filter

The RWBVS comprises two 50-percent supply AHUs, cooling coils to provide normal ventilation and building temperature control. The RWB ventilation exhaust system consists of two 50-percent particulate exhaust filter units each with moisture eliminator, prefilter, electric preheater, absolute (HEPA), non-credited carbon adsorber, post filter (HEPA), ducts and valves, and a fan. The system conforms to RG 1.140 for the filtration unit during normal operation as identified in CESSAR-DC Table 9.4-6. The carbon filter media will conform to Nuclear Grade as defined by the Institute for Nuclear Science. The radiological consequences resulting from gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report.

The particulate and iodine radiation detectors sample the air in ductwork, which serve potentially occupied areas where the potential for the release of radiation exists, and in the exhaust duct header upstream of the filter units. Radioactivity above allowable limits will be indicated and alarmed in the control room and alarmed locally. Upon detection of radioactivity above the allowable limit from the air exhaust, the bypass dampers will be manually closed and the filter units' inlet and outlet dampers will be manually opened to allow the air exhaust filtration. The filtration exhaust fans discharge to the plant vent.

The system is designed to maintain temperature between 4.4 °C and 37.8 °C (40 °F and 100 °F). The system description, components, design parameters, and flow diagram are given in CESSAR-DC Section 9.4.3, Tables 9.4-1, 9.4-3, and 11.3-2, and Figure 9.4-9, respectively. The classification of systems, structures,and components is provided in CESSAR-DC Table 3.2-1 for the RWBVS. The safety-related equipment, fans, dampers, coils and ductwork will be designed and tested in accordance with ASME/ANSI N509, N510, and AG-1 codes and standards.

stray mark

In order to comply with GDC 60, "Control of Release of Radioactive Materials to the Environment," the system needs to conform to RG 1.140. Therefore, the RWB ventilation exhaust system high-efficiency particulate air (HEPA) filters should conform to RG 1.140, Positions C.1 and C.2. This was identified as an DSER Open Item 9.4.3-1 in the DSER. ABB-CE stated in CESSAR-DC Section 9.4.3.1.1.H that the RWBVS will conform to the guidance of RG 1.140. Therefore, DSER Open Item 9.4.3-1 is resolved.

As revised in response to staff RAI Q210-1, ABB-CE stated (in CESSAR-DC Table 3.2-1): (1) radwaste building ventilation system components are non-nuclear Safety Class, non-seismic, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are not applicable; and (2) the radwaste facility structure is seismic Category II as discussed in Chapter 3 of this report, and Appendix B quality assurance requirements are applicable. The staff evaluates flood protection, protection against internally and externally generated missiles, and protection against high- and moderate-energy pipe breaks in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this DSER.

In CESSAR-DC Section 9.4.3, ABB-CE described the system but did not include design parameters and flow diagrams for system components. In the DSER, the staff stated that ABB-CE should provide this information so the staff can complete its review of the system. This was identified as Open Item 9.4.3-2 in the DSER. ABB-CE provided RWBVS flow diagram for system components in the CESSAR-DC Figure 9.4-9 and stated in Section 9.4.3.2.1 that the RWBV supply system consists of two 50 percent capacity supply fans and the exhaust system consists of two 50 percent capacity particulate filtration exhaust units. The components, as shown in CESSAR-DC Figures 9.4-1 and 9.4-9, include a non-credited carbon adsorber and exhaust fans. ~~The staff will evaluate the RWBVS air flow and cooling water design data for the system components during plant-specific review of ABB-CE referencing the System 80+ design.~~ Therefore, DSER Open Item 9.4.3-2 is resolved.

however The design data was not provided does not have any safety significance.

This "states" that NRC review not needed since it is not related to a safety function.

Air flow rates of fans, operating status of fans, temperatures and flow rate of chilled water, damper positions/alignment, air flow rates of supply and exhaust units and air temperatures of supply ventilation units are monitored and indicated in the control room. The pressure drop across the supply filters and exhaust filtration trains is monitored and indicated locally as well as at the radwaste control panel.

See note on pg 107 too!

The system is not safety related, performs no safety-related function for safe shutdown or post accident operation, and failure of the system does not affect the function of other safety-related equipment. Thus, the staff concludes that the RWBVS meets the acceptable criteria of SRP Section 9.4.3 and is, therefore, acceptable pending incorporation of the following item in Amendment V to CESSAR-DC:

1. Revise CESSAR-DC Section 9.4.3 to state that the RWBVS design data for heat load, air, and cooling water for the system components will be provided by a COL applicant for NRC review. This is part of FSER Confirmatory Item 1.1-1.

NRC Management meeting agreed this was NOT required.

9.4.4 Diesel Building Ventilation System

The staff reviewed the diesel building ventilation system (DBVS) in accordance with SRP Section 9.4.5 (NUREG-0800). The design has two redundant emergency diesel generator (EDG) located in separate areas inside the nuclear annex on opposite sides of the reactor building. Each EDG area is served by a ventilation system designed to maintain acceptable environmental conditions for operation, testing, and maintenance of the equipment, and to allow for personnel access.

The DBVS is designed for once-through flow using inlet and exhaust fans, filters, and dampers. The system is designed to maintain temperature between a minimum of 4.4 °C (40 °F) and a maximum of 49 °C (120 °F) when the DG is not operating and between a minimum of 4.4 °C (40 °F) and a maximum of 50 °C (122 °F) when the DG is operating. Electric heaters, activated on low temperature, maintain temperature above freezing and fans are automatically activated to control elevated temperature. Air intake structures and exhaust vents are protected against the effects of natural phenomena and missiles.

Each division of non-safety-related supply portion of the system consists of one 100-percent-supply fan equipped with damper and prefilter. Air is exhausted to the outdoors through each division of the safety-related exhaust portion of the system which consist two 50-percent-supply fans. Each fan is equipped with a two speed motor and has a separate exhaust vent. The system identify

In the DSER, the staff stated that CESSAR-DC Table 3.2-1 should appropriately ~~upidentify~~ the system, system components, and their locations with respect to safety class, seismic Category, and quality assurance requirements designations. This was identified as DSER Open Item 9.4.5-1 in the DSER. Subsequently, ABB-CE provided the requested information identifying the exhaust system as seismic Category I, Safety Class 3 and Quality Class 1, and the supply system as seismic Category II, non-nuclear safety class and Quality Class 2, except heating and cooling coils which are non-seismic, non-nuclear safety class for the Quality Class 3. Therefore, DSER Open Item 9.4.5-1 is resolved.

Outdoor air is drawn into the non-safety-related ventilation supply system serving each division through one 100-percent capacity supply unit consisting of a prefilter and cooling/heating coils by two 100-percent-supply fans. The fresh air intake structures are located in the control areas duct shaft and are protected against such environmental conditions as high winds, rain, snow, and ice. The supply fans and conditioning unit are not safety-related units. Supply air is distributed to equipment rooms and access areas in the subsphere building and exhausted from the building through a filtration unit by two 100-percent capacity exhaust fans. The filtration unit and exhaust fans are safety-related equipment. The fans are powered from a Class 1E supply, backed up by the emergency DG.

Originally, ABB-CE did not provide information regarding the intake air vents conformance with GDC 17 requirement as it relates to assuring proper functioning of the safety-related equipment, except for mentioning that the air is filtered. This was identified as DSER Open Item 9.4.5-2 in the DSER. SRP Section 9.4.5 provides guidance to ensure that adequate means is provided in the system design for control of airborne particulate material (dust) accumulation. The system arrangement should provide a minimum of 6.1m (20 ft) from the bottom of the fresh air intakes to grade elevation.

Subsequently, ABB-CE provided above information in CESSAR-DC Section 9.4.5.3, stating that the fresh air intakes are located at least 9.14m (30 ft) above grade elevation (as shown in CESSAR-DC Figure 1.2-8) to minimize intake of dust into the building and are provided with tornado dampers. Therefore, DSER Open Item 9.4.5-2 is resolved.

The divisional exhaust filtration unit consists of a moisture eliminator, prefilter, electric preheater, ~~non~~ Non-Credited carbon bed adsorber, and absolute and post filters (HEPA) upstream and downstream of the carbon adsorber as shown in CESSAR-DC Figures 9.4-1 and 9.4-5. A motor-operated damper on the downstream of exhaust fans is for tornado protection and for isolation when exhaust fans are off. CESSAR-DC Table 9.4-3, "Input for Release Analysis Filter Efficiencies," shows the creditable HEPA efficiency of 99-percent for post-accident releases. ABB-CE stated in the CESSAR-DC Section 9.4.5.3 that the dose analysis, to support 10 CFR Part 100 limits following a LOCA or DBA, only takes credit for the HEPA filters in the filter train and no credit is taken for the ~~charcoal~~ Carbon adsorbers.

The staff concluded in Section 15.A.11, that with respect to the radiological consequences of all potential accidents, credit is given for the removal of particulate iodines only. Therefore, ~~charcoal~~ Carbon adsorbers need not be credited in the SBVS.

In addition to the air supply and filtration function, each divisional system includes separate individual safety-related cooling units for each of the equipment rooms. The safety-related equipment includes containment spray pumps and heat exchangers, safety injection system pumps and heat exchangers, shutdown cooling system pumps and heat exchangers, fuel pool heat exchangers, motor and steam-driven emergency feedwater pumps, and penetration rooms. The safety-related cooling units recirculate air through prefilters, cooling coils serviced from the safety-related CWS, and fans. The safety-related equipment room AHUs are powered by a Class 1E source, backed up by the emergency DG. All cooling units are started automatically and remain operational throughout a LOCA event. All safety-related system components are designed to permit in-service inspection. The safety-related equipment room cooling units are designed to maintain the space temperature below 38 °C (100 °F). At least one train of safety-related equipment rooms is maintained below 38 °C (100 °F) assuming a single failure of an active component concurrent with an LOOP.

safety-related subsphere ventilation system components are designed to permit inservice inspection; (2) the failure of the non-safety-related supply fan has no effect on the exhaust fan since negative pressure is maintained inside the subsphere and the exhaust is filtered; (3) the fresh air intakes are located at least 9.14m (30 ft) above grade elevation and protected against adverse environmental conditions; (4) a HEPA filter is provided downstream of the carbon adsorbers and filtration components in each filtration train are shown in accordance with CESSAR-DC Figure 9.4-1 to satisfy RG 1.52; and (5) the system includes differential pressure alarms and indication in conformance with the guidance of RG 1.140, as referenced in CESSAR-DC Sections 9.4.5.1 and 9.4.5.3. Therefore, Confirmatory Item 9.4.5-1 is resolved.

The staff concludes that the SBVS complies with the applicable GDC referenced in SRP Section 9.4.5 and is, therefore, acceptable pending incorporation of the following items (part of FSER Confirmatory Item 1.1-1) in Amendment V to CESSAR-DC:

1. Revise CESSAR-DC Section 9.4.5.3 on Page 9.4-28 to state that the HEPA filters are designed to limit the offsite dose within the requirements of 10 CFR Part 100. Also, revise CESSAR-DC Section 9.4.5.1 on Page 9.4-24 to state that the SBVS is designed to limit the offsite dose following a LOCA or DBA within the requirements of 10 CFR Part 100 and delete reference of SRP 6.4.

2. Add in the end of first paragraph on CESSAR-DC Page 9.4-28 to state that, "The ductwork from the building exit up to and including the isolation damper are qualified for the tornado differential pressure."

9.4.6 Containment Cooling and Ventilation System

The staff reviewed the containment cooling and ventilation system (CC&VS) in accordance with SRP Section 9.4.5 (NUREG-0800). This system maintains suitable environmental conditions inside the containment for normal operation, maintenance, and testing. The system is not safety related except for dampers and penetration ductwork that isolate portions of the system inside the containment from portions of the system located in the nuclear annex. The low purge and high purge systems are designed to maintain the containment under slight negative pressure with respect to the atmosphere.

The CC&VS is comprised of: (1) the recirculation cooling system; (2) the low- and high-purge supply and exhaust subsystems; (3) the containment air cleanup system; (4) the pressurizer compartment cooling redundant fans; (5) the reactor cavity compartment cooling redundant fans; and (6) the CEDM cooling system.

The recirculation cooling system consists of four 33 percent capacity recirculation cooling units. The recirculation cooling units remove heat in the containment, generated by the nuclear steam supply system support structures and RCS insulation heat loads (SSAR Tables 9.4-4 and 9.4-2), and maintain the served areas between 15.5 °C and 43.3 °C (60 °F and 110 °F).

The low-purge subsystem relieves containment pressure during startup and shutdown. In-containment refueling water storage tank (IRWST) purge supply and exhaust are normally closed and opened only for personnel access. The high-purge system operates to reduce radiation levels before and during personnel access to the containment. The containment high-purge system mitigates the radiological consequences of a postulated fuel-handling accident inside containment to conform with 10 CFR Part 100 requirements and is not used during power operation.

The containment air cleanup system consists of prefilter, absolute HEPA filter, carbon adsorber, post HEPA filter and a fan. It is designed to reduce containment airborne concentrations to approximately seven maximum permissible concentrations (MPC) to permit personnel access and conforms to ANSI/ANS-56.6, "Pressurized Water Reactor Containment Ventilation Systems."

The reactor cavity compartment cooling and pressurizer compartment cooling fans, in conjunction with the recirculation cooling system, maintain the served areas below 54.4 °C (130 °F).

The CEDM cooling system consists of redundant cooling units. The CEDM cooling system maintains the served areas to 76.6 °C (170 °F).

recirculation fans are provided to create the uniform mixture, as required. Also, radiation monitoring information is provided in CESSAR-DC Section 11.5. TBVS removes the heat dissipated by equipment, piping, lighting and solar heat gains, and maintains the served areas between 4.4 °C and 43.3 °C (40 °F and 110 °F). The design outside temperature will be based on the 5 percent exceedance air temperature values. The system description and layout drawings are given in CESSAR-DC Section 9.4.7 and Figures 1.2-18 and 1.2-19, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the TBVS. the The design parameters and flow diagram ~~will be reviewed and evaluated during plant specific review for ABB-CE referencing the System 80+ design.~~ Therefore, DSER Open Item 9.4.7-1 is resolved.

Because this is a non safety system

do not have any safety significa

The system instrumentation for manual and automatic operations and system verification is provided locally. The fan indications and alarms are also provided locally.

The review established that the system is not safety related and that failure of the system does not compromise the operation of safety-related systems. Therefore, the requirements of GDC 2, 4, 17, and 60 are not applicable to this system.

The staff concludes that the TBVS complies with the acceptance criteria of SRP Section 9.4.4 and is, therefore, acceptable pending satisfactory resolution of the following discrepancy:

1. Revise CESSAR-DC Section 9.4.7 to state that the NRC will evaluate the design parameters for TBVS system components and flow diagram during plant-specific review for ABB-CE referencing the System 80+ design.

9.4.8 Station Service Water Pump Structure Ventilation System

In response to staff RAI Q410.121, ABB-CE stated that the station service water pump structure ventilation system (SSWPSVS) is dependent on site-specific considerations. ABB-CE committed to provide interface requirements for the station service water pump structure ventilation system. This was identified

fied as Confirmatory Item 9.4.9-1 in the DSER. Subsequently, ABB-CE stated that: (1) the heat loads and design parameters are provided in Table 9.4-1; (2) the physical location of major components is shown in Figure 1.2-8 and the system flow diagram is provided in Figure 9.4-8; (3) all safety-related components are designed to permit inservice inspection as stated in CESSAR-DC Section 9.4.9.4; (4) the nuclear annex structure is designed to seismic Category I standards; and (5) an LOOP will not affect the safety function of safety-related equipment. Therefore, Confirmatory Item 9.4.9-1 is resolved.

The staff concludes that the NAVS complies with the applicable GDC referenced in SRP Section 9.4.5 and is, therefore, acceptable pending incorporation of the following confirmatory item in Amendment V to CESSAR-DC:

1. Revise CESSAR-DC Section 9.4.9.2.1 to state that the isolation dampers are manually closed during a tornado warning. This is part of FSER Confirmatory Item 1.1-1.

9.4.10 Component Cooling Water Heat Exchanger Structure(s) Ventilation Systems

The staff reviewed the component cooling water heat exchanger structure(s) Ventilation Systems (CCWHXSVS) in accordance with SRP Section 9.4.5 (NUREG-0800). As identified in the CESSAR-DC Table 3.2-1, the CCWHXSVS components are located completely within seismic Category I structures, and fans, dampers, and ductwork are protected from floods and tornado missile damage and interaction with other non-seismic systems. The fans, dampers, ductwork, unit heaters and supports are designed as seismic Category II, non-nuclear safety class, and the quality assurance requirements of 10 CFR Part 50, Appendix B do not apply. The system is not required to operate for the CCWS to perform its safety function. ~~The component~~ cooling water heat exchanger structure(s) is seismic Category I, Nuclear Safety Class 3, and the quality assurance requirements of 10 CFR Part 50, Appendix B apply.

not needed in this section

This # addresses ventilation, not structures

divisions

Two CCWHXSVS are provided, one for each division of CCW. The two systems are physically separated and there is no interaction between the systems. Each system consists of a fan, associated motor operated intake and exhaust

division

dampers, ductwork, supports and instrumentations and controls. The system description, design parameters, and flow diagram are given in CESSAR-DC Section 9.4.10, Table 9.4-1, and Figure 9.4-10, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the CCWHXSVS.

The component flow design parameters will be reviewed and evaluated during plant specific review for ABB-CE referencing the System 80+ design.

The system maintains the served areas above 4.4 °C (40 °F). The system fresh air intakes are located at least 6.1m (20 ft) above grade elevation and away from plant discharges to minimize intake of dust and contaminants into the structures.

The CCWS fluid is monitored by radiation detectors and any radioactivity present is contained within the piping system, therefore, no provisions are made to contain the release of the radioactive materials in the CCWHXSVS.

The system instrumentation for manual and automatic operations and system verification is provided remotely or locally. The fan indications and alarms are provided in the control room. The space temperature indication and high/low alarms are also provided in the control room.

The review established that the system is not safety related and that failure of the system does not compromise the operation of safety-related systems. Therefore, the requirements of GDC 2, 4, 17, and 60 are not applicable to this system.

The CCWHXSVS complies with the acceptance criteria of SRP Section 9.4.5 and is, therefore, acceptable pending incorporation of the following confirmatory item in Amendment V to CESSAR-DC:

1. Revise CESSAR-DC Figure 9.4-10 to include fan status. This is part of FSER Confirmatory Item 1.1-1.

offsite power. The discussion addresses how the performance of these safe shutdown functions will not be compromised by spurious operations induced by a fire either inside or outside the containment. Specifically, the subject section indicates that adverse effects due to fire induced spurious operations will be prevented by one, or an applicable combination, of the following design features: (1) needed shutdown system lines will have two power operated valves in series with the valves powered by different divisions or different channels within a division.~~and the valves will be widely separated and in different fire areas;~~ (2) the associated MCCs for the valves will be in different fire areas; and (3) the MCC breakers associated with the valves are opened once the valves are placed in desired position (i.e., closed or open). The section also states that the solenoid valve power supply fuses are normally removed to prevent fire induced spurious opening of the single isolation valves provided on each of the two vent lines of each safety injection tank. ABB-CE has also provided a Fire Hazards Assessment document to the NRC. This document, among other things, includes a safe shutdown analysis for System 80+. In the document, ABB-CE has listed or discussed, as appropriate, the following: (1) the criteria for achieving and maintaining safe shutdown following a fire (i.e., the ability to achieve and maintain safe shutdown without entering into the fire area for repairs or manual operations); (2) design basis goals for safe shutdown; (3) safe shutdown performance objectives; (4) systems required for safe shutdown; (5) safe shutdown components; (6) protection against associated circuit concerns; (7) prevention of fire-induced high/low pressure interface breaches; and (8) a list identifying fire areas that contain equipment required for safe shutdown following a fire and the redundant areas that contain the corresponding redundant equipment. Regarding preventing fire-induced high/low pressure interface breaches, the subject document (Section 7.6) states that the RCS MOVs which serve as high/low pressure interfaces and are required to be closed during normal power operation, will have the valve motors deenergized during power operation to prevent such fire-induced breaches.

Based on its review of the Fire Hazards Assessment document and CESSAR-DC subsections 9.5.1.3.6, 9.5.1.3.7 and 9.5.1.3.8, the staff concludes that associated circuit interactions due to a fire in any plant fire area will not divisions, and, with two exceptions, HVAC ducting does not penetrate three-hour-rated fire barriers separating divisions. ~~Therefore, with two exceptions, fire dampers are eliminated from the ABB-CE System 80+ design. This simplifies not only the design of the System 80+ HVAC systems, but also installation and maintenance of the system throughout the life of the plant.~~ Fire Dampers are still provided between fire areas within a division

One exception to the division-specific HVAC system is a single opening in the divisional fire wall that separates the redundant AHUs. An air intake duct that supplies makeup air to the redundant control room system passes through this opening. This arrangement, which is necessary for nuclear safety reasons, enables makeup air to be drawn from either side of the facility. The opening is protected with a combination fire and smoke damper. The other exception is the fuel building ventilation system.

In the DSER, the staff identified the need for a description of the design and operation of the components used in the smoke removal mode of operation. This was identified as DSER Open Item 9.5.2.2-2 in the DSER.

In the CESSAR-DC, Section 9.5.1.2, ABB-CE indicated that the HVAC system will be designed to remove smoke and mitigate smoke migration beyond the area of origin in the event of fire. The dedicated fans for smoke purge will be designed to exhaust at a minimum of 945 L/min per m^2 (3 CFM/ft^2) of floor area. The normal ventilation is designed to provide an air flow of 315 L/min per m^2 (1 CFM/ft^2) of floor area or more. ABB-CE indicated that the layout of the ductwork is such that it ensures ventilation of all corners of the area as much as practical. The design as described will provide a lower pressure into the division experiencing the fire that will prevent or significantly reduce the amount of smoke migration to other divisions. In CESSAR-DC, Amendment U, Section 9.5.1.8.2, ABB-CE indicated that the ventilation system will be designed in accordance with NFPA 92B, "Guide for Smoke Management Systems in Malls, Atriums and Large Areas." ABB-CE's proposed HVAC design is in accordance with BTP CMEB 9.5-1 and SECY-90-016 and is acceptable. Therefore, DSER Open Item 9.5.1.2.2-2 is resolved.

ABB-CE's commitment to design the system in accordance with National Fire Protection Association Standards is in accordance with BTP CMEB 9.5-1 and is, therefore, acceptable.

2. The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings must be designed in compliance with ANSI B31.1 and analyzed to remain functional following a safe-shutdown earthquake. A portion of the water-supply system, including a tank, a pump, and part of the yard supply main must be designed to these requirements also. The remainder of the water systems must be designed to the appropriate fire-protection standards. During normal operation, the seismically designed and non-seismically designed systems must be separated by normally closed valves and a check valve, so that a break in the non-seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system.

CESSAR-DC Section 9.5.1.7.3.C indicated that the sprinkler system piping is seismically restrained to avoid interaction with systems, equipment, and components which must function following a design basis seismic event. Also, Section 9.5.1.7.4 stated that "fire hose and standpipe systems located in the Reactor Building and Nuclear Annex meet Seismic Category I requirements ~~are designed to be operational following the DBE~~. Each connection of the standpipe system to the fire protection water distribution system includes a manual isolation and a back flow prevention check valve which is seismically qualified."

The fire hose and standpipe systems located in the Reactor Building and the nuclear annex will be designed to remain functional following a safe shutdown earthquake. The piping system serving such hose stations will be analyzed for SSE loading and will be provided with supports to ensure system pressure integrity. The piping and valves for the portion of hose standpipe system affected by this functional requirement will be designed, as a minimum, to satisfy ANSI B31.1. The System 80+ design, as discussed, is in accordance with the BTP CMEB 9.5-1 and, therefore, is acceptable.

"Air Conditioning and Ventilation Systems"

CESSAR-DC Section 9.5.1.8.4 states that where fixed fire protection systems are installed, floor drains are provided, sized to collect water discharge. In areas where drains are not installed due to pressure boundary constraints, equipment susceptible to water damage is installed on six inch elevated curbs. Based on the above, DSER Open Item 9.5.1.4.3-1 is resolved.

9.5.1.4.4 Smoke Control

The DSER stated that ABB-CE must submit more detailed information on utilization of the HVAC system for smoke removal and control during fire. This was identified as DSER Open Item 9.5.1.4.4-1. CESSAR-DC, subsection 9.5.1.8.2 indicated that the ventilation systems will be designed in accordance with NFPA 90A and NFPA 92B, "Guide for Smoke Management Systems in Malls, Atriums, and Large Areas." As discussed in Section 9.5.1.2.2 of this SER, ABB-CE's response is acceptable. Therefore, DSER Open Item 9.5.1.4.4-1 is resolved.

9.5.1.4.5 Access/Egress Routes

Section 1.4.1 of the ABB-CE System 80+ Design Fire Hazards Assessment, states that the plant arrangement is carefully evaluated to ensure adequate means of personnel egress and fire brigade access are provided. Additionally, in Section 3.2 of the Fire Hazards Assessment, ABB-CE states that it will comply with the provisions of SRP Section 9.5.1. The staff accepts this as a commitment to provide clearly marked exit routes for each fire area. These routes will be designed to comply with applicable life safety codes and standards. These provisions for access and egress routes conform to the guidelines in Section C.5.g of BTP CMEB 9.5-1 and Section III.G of Appendix R to 10 CFR Part 50, and are acceptable.

9.5.1.4.6 Construction Materials and Combustible Contents

ABB-CE has committed in the System 80+ Design Fire Hazards Assessment Section 1.4.1 to furnish appropriate fire-resistance ratings for structural members, and noncombustible or fire-retardant interior finish materials. ABB-CE also committed in SRP Section 3.2 to comply with the provisions of SRP Section 9.5.1.

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fire barriers for areas as indicated by the BTP CMEB 9.5-1. ABB-CE commits to meet BTP CMEB 9.5-1 (and SECY-90016) for the installation of fire protection features. Therefore, DSER Open Item 9.5.1.6-4 is resolved.

5. The normal HVAC system will be utilized for smoke removal from any area with a fire, and for smoke control to prevent migration from an area with a fire to other fire areas. The DSER indicated that ABB-CE has not submitted details of operation of the HVAC system operating in the smoke control/smoke purge mode. Therefore, the staff is unable to complete its review of this mode of operation of the System 80+ design HVAC system. This was identified as DSER Open Item 9.5.1.2.2-2 in the DSER.

   As previously discussed in Section 9.5.1.2.2 of this SER, ABB-CE indicated that the HVAC system will be designed to remove smoke and mitigate smoke migration beyond the area of origin in the event of fire. The dedicated fans for smoke purge will be designed to exhaust at a minimum of 945 L/min per m^2 (3 CFM/ft^2) of floor area. The normal ventilation is designed to provide an air flow of 315 L/min per m^2 (1 CFM/ft^2) of floor area or more.

   In the CESSAR-DC, Amendment U, Section 9.5.1.8.2, ABB-CE further indicated that the ventilation system will be designed in accordance with NFPA 92B, "Guide for Smoke Management Systems in Malls, Atriums and Large Areas." ABB-CE's proposed HVAC design is in accordance with CMEB BTP 9.5-1 and SECY-90-016 and is acceptable. Therefore, DSER Open Item 9.5.1.2.2-2 is resolved.

6. ABB-CE must confirm that no penetrations exist in the three-hour-rated barriers separating fire areas containing redundant trains of safe-shutdown equipment. This was identified as DSER Open Item 9.5.1.6-5 in DSER.

   In a letter dated June 11, 1993, ABB-CE indicated that with few exceptions, there are no openings in the three-hour-rated wall between redundant equipment required for safe shutdown. In cases where there

Compliance of the System 80+ DG auxiliary support systems with the recommendations of NUREG/CR-0660 is summarized in Table 9.1.2 to this FSER. Compliance to individual recommendations is discussed in other sections of this report concerned with applicable DG auxiliary support systems.

Security Considerations

The staff considers that the DG and its support systems (fuel, cooling water, starting air, lube oil, exhaust, field flashing, and instrumentation and controls) are vital systems; therefore, as required by 10 CFR 73.55(c), access to all DG and vital support systems' components should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desired, if access to each vital area is otherwise controlled.) The description in CESSAR-DC Sections 9.5.5, 9.5.6, 9.5.7, and 9.5.8 of the protected location of DG components (i.e., the DG building) did not address protection from sabotage. The DG building was not included as a vital area in ABB-CE's response of September 28, 1989, to RAI Q500.7; also, ABB-CE's response to followup RAI Q500.21 referred to the DG building as a vital area only in the sense of radiation protection guidance of RG 1.97 and NUREG-0737. Vital designation of the DG system in the sense of 10 CFR Part 73 was identified as DSER Open Item 9.5.4.1-1. Subsequently, ABB-CE revised CESSAR-DC Sections 9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8 and 9.5.9 to designate the DG's systems as vital systems. This change resolved the staff's concerns about designation of the DG systems and this item is considered closed.

In a letter of February 28, 1992, EPRI advised ABB-CE to add Section 9.5.4.1.2 ~~9.5.1.4.2~~ to the CESSAR-DC stating that the diesel fuel storage structure is a seismic Category I structure within the scope of the operating license applicant and requiring ABB-CE to build a structure that is in the "vital protection area" and that will "withstand the effects" of a sabotage event. The staff interprets that terminology to mean that access to equipment within the diesel fuel storage structure requires passage through both the protected area fence and an additional vital barrier. The proposed vital area designation is in accordance with the NRC Review Guideline 17 criterion that seismic Category I equipment be considered vital equipment and is acceptable. However, the proposed description of the ventilation system for the diesel fuel storage level alarm, indicating that 60 minutes of fuel remains at full load, allows the operator to manually open a valve on the bypass line to fill the day tank. In the event the transfer valve remains in the open position, fuel oil would continue to flow from the storage tanks to the day tank until the system reached hydrostatic equilibrium. Since there is no day tank overflow line, fuel oil would rise in the day tank vent pipe to a level that is equivalent to level of the fuel oil in the storage tanks but well below the top of the vent.

All piping and components are located in seismic Category I buildings, except for a portion of the ANSI Class 3 seismically qualified piping from the fuel oil storage tanks to the day tank. The external surfaces of carbon steel tanks and underground components are coated for corrosion protection. In addition to being coated, the external surfaces of buried piping and tanks are protected from corrosion by an impressed-current cathodic protection system in accordance with NACE Standard RP-01-69, or other methods deemed appropriate based on site-specific conditions. The diesel engine and engine-mounted components are constructed in accordance with IEEE Standard 387, "IEEE Standard Criteria For Diesel Generator Units Applied As Standby Power Supplies For Nuclear Power Generating Stations." The DGFOSTS is designed and constructed in compliance with ANSI Standard N195, "Fuel Oil Systems for Standby Diesel Generators," except in regard to the flame arresters on the storage tanks, which are not required specifically by SRP Section 9.5.1, an overflow line from the day tank, and excluding all references to the fuel oil transfer pumps.

The System 80+ design utilizes two half-capacity fuel oil storage tanks per diesel which provides the ability to operate the diesel off one tank while isolating and filling the other tank. After filling a storage tank, a period of not less than 24 hours is required to allow any stirred sediment to settle before aligning that tank to its respective diesel. The filtering and recirculation process is performed on a tank-by-tank basis, with the frequency of operation dependent on the results of a fuel oil inspection program. Accumulated water in the fuel oil storage tanks will be removed by the recirculation system through a sample connection provided on the recirculation pump dis-

COMMENTS ON FSER

The following are comments on individual FSER sections. In addition, markups of these sections are provided.

9.2.1 Station Service Water System

1. **(Refer to FSER page 9-54)**

   The SSWS sump pumps are Safety Class NNS and non-seismic. See Table 3.2-1.

2. **(Refer to FSER page 9-55)**

   INSERT A: (from CESSAR-DC Section 9.2.1.2.1.1)

   "the highest expected operating temperature and flow, at the normal water elevation, and assuming the traveling screens are 50% clogged. The available NPSH exceeds the required NPSH for worst case water elevations for all operation, flow, and temperature conditions."

3. **(Refer to FSER page 9-55)**

   INSERT B: (from CESSAR-DC Section 9.2.1.2.1.1)

   "(Note: For worst case UHS water elevation, the margins previously specified need not apply.)"

9.2.2 Component Cooling Water System

4. **(Refer to FSER page 9-59)**

   INSERT C: (from CESSAR-DC Section 9.2.2.1.1 (F))

   "Containment isolation valves and containment penetration piping are Seismic Category I and Safety Class 2."

5. **(Refer to FSER page 9-59)**

   Diesel generator engine starting air aftercoolers are non-essential. See CESSAR-DC Section 9.2.2.2.

COMMENTS ON FSER

6. **(Refer to FSER page 9-59)**

<u>**INSERT D:**</u> (from CESSAR-DC Section 9.2.2.2)

"Non-essential components are supplied component cooling water by means of non-nuclear safety class cooling loops with the exception of the charging pump motor coolers and miniflow heat exchangers, the instrument air compressors, and the diesel generator engine starting air aftercoolers which are supplied component cooling water by means of Safety Class 3 cooling loops."

7. **(Refer to FSER page 9-59)**

There are valves in the CCWS (i.e., CC-145 and CC-245 on the discharge line of the charging pump miniflow heat exchangers) that utilize instrument air but are not safety related. It is not necessary for these valves to have safety-grade operators and solenoid valves.

8. **(Refer to FSER pages 9-61 through 9-66)**

Pages 9-61 through 9-66 are in the FSER twice.

9.2.5 Ultimate Heat Sink

SEE MARKUPS.

9.2.9 Chilled Water System

9. **(Refer to FSER page 9-73)**

The computer room is supplied by the NCWS.

10. **(Refer to FSER page 9-76)**

Air-operated butterfly valves are shown in CESSAR-DC Figure 9.2.9-1.

9.3.1 Compressed Air Systems

SEE MARKUPS.

COMMENTS ON FSER

9.3.3 Equipment and Floor Drainage System

NO COMMENTS.

9.5.2 Communication Systems

NO COMMENTS.

9.5.3 Lighting System

11. **(Refer to FSER page 9-169)**

<u>**INSERT E:**</u>

"The security lighting system will be considered part of the permanent non-safety systems and will be fed from the Alternate AC (AAC) Source (Combustion Turbine). Selected portions of the security lighting system essential to maintaining adequate plant protection are powered from a non-Class 1E battery power source."

12. **(Refer to FSER page 9-171)**

This information is incorrect. The Class 1E distribution system does **not** supply at least one of the circuits supplying the lighting fixtures for the normal lighting system in safety related areas. The following is a brief explanation of the normal lighting system and the emergency lighting system:

> The normal lighting system provides general illumination throughout the plant. The circuits to the individual lighting fixtures are staggered as much as possible with the staggered circuits fed from separate electrical divisions to ensure some lighting is retained in the room in the event of a circuit failure. The normal lighting system is considered part of the plant permanent non-safety systems. As such, the normal lighting system is energized as long as power from an offsite power source or a standby non-safety source (Combustion Turbine) is available. The Combustion Turbine is designed to start automatically within two minutes from the onset of a LOOP event.

The emergency lighting system is used to provide acceptable levels of illumination in vital areas throughout the plant upon loss of the normal lighting system. Emergency lighting is accomplished by conventional AC fixtures fed from Class 1E AC power sources and Class 1E DC self contained, battery operated lighting units. Class 1E DC self contained, battery operated lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located.

Emergency lighting in the main control room is provided such that at least two circuits of lighting fixtures are powered from different Class 1E divisions. The emergency lighting system in the main control room maintains minimum illumination levels in the main control room during emergency conditions including station blackout.

13. **(Refer to FSER page 9-172)**

INSERT G:

"the luminaries are of a proven design with long life and low maintenance requirements, such as fluorescent, metal-halide, and high pressure sodium lamps. Mercury vapor lamps are not used in fuel handling areas."

14. **(Refer to FSER page 9-170)**

Emergency procedures and hazard analyses have not been completed. They will be completed as COL applicant items.

9.5.10 Compressed Gas System

NO COMMENTS.

The SSWS pumps are located in a seismic Category I structure that is protected from floods and tornado missiles. The SSWS pumps, strainers, ~~sump pumps,~~ and traveling screens are seismic Category I, Safety Class 3, Quality Class I, as is shown in CESSAR-DC Table 3.2-1. In addition, the SSWS ~~is designed~~ to preclude any adverse interaction with non-seismic systems in the vicinity. Therefore, the design presented in the CESSAR-DC satisfies GDC 2 by meeting the guidance of RG 1.29, Position C.1, with respect to its seismic requirements.

All essential SSWS components are fully protected from floods, tornado-missiles, internal missiles, pipe breaks, pipe whip, and jet impingement. In addition, the system is designed to minimize the potential for water hammer by providing for adequate filling and high-point vents. The SSWS is also installed underground or in buildings that will protect it from adverse environmental conditions. In the event of a loss-of-offsite power (LOOP), the SSWS will be shut down and restarted in accordance with the diesel generator (DG) load sequencing. The DG sequencing times are confirmed to be commensurate with SSWS requirements regarding component cooling. Accordingly, the design presented in the CESSAR-DC satisfies GDC 4.

The staff reviewed the design of the SSWS to identify shared systems and components. The two divisions of the SSWS are physically and electrically separate and share no components or systems. Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant would have 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 structure, system, and components (SSCs) to perform their required safety functions. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units and, therefore, meets the requirements of GDC 5.

Each division can provide safety-grade shutdown cooling via both pumps for up to 36 hours and post-loss of coolant accident (LOCA) cooling via one pump for up to 30 days. Each of the four identical SSWS pumps (two per division) can provide 100 percent of the required flow for post-LOCA conditions. During normal operation, only one pump per division is required to be operating. If a low pump discharge signal is received, the second pump in the respective division automatically starts. The pumps are of the vertical centrifugal type and are installed so that they meet the minimum net positive suction head (NPSH) at ~~the simultaneous occurrence of UHS pond draw-down, maximum pond temperature, maximum flow through the screens and piping to the pits, and the assumption that the safety-grade screens are clogged.~~ ADD INSERT A The minimum available NPSH is the smaller of either 25 percent ~~of~~ greater than, or 3m (10 ft) greater than, the required NPSH specified by the pump vendor. ADD INSERT B The pumps have at least a 7-percent margin in head at the pump design point. The head versus flow curve is continuously rising from the design point to shutoff.

Instrumentation that monitors the SSWS flow, temperature, and system pressure, as well as radiation levels within the SSWS inventory, supports automatic system actuation features and alarms to alert the operator to anomalous operating conditions. These features ensure that the SSWS is properly removing heat from the CCWS and transferring heat to the UHS. In addition, these features detect pipe breaks and related system failures to minimize the resulting adverse consequences and to prompt mitigating actions. As noted earlier, the design comprises two full-capacity divisions, each of which has two redundant trains to provide the necessary cooling. The system is designed to accommodate a single failure in a train and compensates for the postulated single failure via: (1) reliance on the redundant train within the division or; (2) the two trains of cooling provided in the other division, or both, depending on the single failure. The staff also reviewed the design to ensure that isolation valves were installed and could be remotely operated to ensure that the system's safety function would not be compromised by a pipe break, a component failure, or a related failure. ABB-CE incorporated adequate isolation and control provisions into the design to protect the system from postulated failures. Therefore, the system meets the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat from SSCs important to the safety of the UHS.

Each division of the CCWS includes the following: two heat exchangers, a
surge tank, two component cooling water pumps, a chemical addition tank, a
component cooling water radiation monitor, two sump pumps, a component cooling
water heat exchanger structure sump pump, and related piping, valves, instru-
mentation and controls. No cross-connection exists between the two divisions.
The CCWS is cooled by the SSWS that removes heat from the tube side of the
CCWS heat exchangers. To preclude leakage from the SSWS to the CCWS, the CCWS
operates at a higher pressure than the SSWS.

ADD INSERT C

Each CCWS division consists of an essential and non-essential cooling loop.
The essential cooling loop piping and components (e.g., heat exchangers, pumps
and surge tanks) ~~CIVs, and containment penetration piping~~ are seismic ④
Category I and Safety Class 3. The essential portion of the CCWS supplies
cooling to the following redundant safety-related components: shutdown
cooling heat exchangers, mini-flow heat exchangers, and pump motor coolers;
safety injection pump motor coolers; containment spray heat exchangers, mini-
flow heat exchangers, and pump motor coolers; component cooling water pump
motor coolers; spent fuel pool heat exchangers and pump motor coolers; motor
driven emergency feedwater pump motor coolers; DG jacket water coolers ~~and~~ ⑤
~~engine starting air aftercooler~~; and essential chillers. The non-essential
portion of the CCWS supplies cooling to the reactor coolant pump motor air
coolers, ~~upper and lower bearing~~ motor oil coolers, oil coolers, seal coolers, and
high pressure coolers; letdown heat exchanger; ~~charging pump motor coolers and~~
~~mini-flow heat exchanger;~~ sample heat exchangers; gas stripper ~~overhead~~ ⑥
~~condenser and aftercooler~~; boric acid concentrator ~~distillate cooler, condens-~~
~~er, and condenser vent cooler~~; normal ~~non-essential~~ chilled water condensers; and
instrument air compressors ~~and aftercoolers~~. The isolation valves separating
seismic Category I portions from the nonseismic portions are Class 2 or 3
(i.e., Quality Group B or C, respectively). ~~The non-essential loops are~~ ⑦
~~composed of non-nuclear safety piping and components, with the exception of~~
~~the CIVs and penetration piping (these are ANSI Safety Class 2)~~. In addition,
the CCWS is designed to preclude any adverse interaction with non-seismic
systems in the vicinity. Therefore, the design presented in the CESSAR-DC
satisfies GDC 2 with respect to its seismic requirements by virtue of meeting

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the guidance of RG 1.29, Position C.1, with respect to safety-related portions of the system, and Position C.2, with respect to non-safety-related portions of the system.

All essential CCWS components are fully protected from floods, tornado-missiles, internal missiles, pipe breaks, pipe whip, and jet impingement. The two divisions of the CCWS are physically separated and are routed so as to be protected from adverse environmental conditions that could impair performance. In addition, in responding to Q410.76, ABB-CE stated that the effects of high- and medium-energy pipe breaks are considered in the design of the CCWS. Specifically, the response indicates: "The CCWS safety-related components are designed and protected such that this type of failure would not affect the safety performance of the CCWS." See Section 3.5 of this SER. Accordingly, as presented in the CESSAR-DC, the design satisfies GDC 4.

The CCWS is also designed to minimize the potential for water hammer by having vents in all high points and drains in all low points of the system. Vents are located to ensure that the piping is filled with water; this reduces the chances of water hammer after pump start up. Also, valve opening and closing times are selected to minimize water hammer effects. Similarly, the system's mechanism for venting, the surge tanks are located at the system's high point to facilitate venting and filling.

The CCWS receives power from the Class 1E auxiliary power system. In the event of an LOOP, DGs provide power to the auxiliary power system. Each of the two DGs is capable of supplying 100 percent of the power required for operating a division of the necessary safety-related equipment. An LOOP would result in the shutdown of the CCWS and its subsequent restart, in accordance with the DG's load sequencing. The sequencing logic ensures that the appropriate CCWS pump is loaded within approximately 10 seconds.

Several valves in the CCWS rely on the compressed air (instrument air) system for operation. In the CCWS, all active safety-related valves dependent on compressed air have safety-grade operators and solenoid valves. In the event of a loss of the compressed air system (e.g., during an LOOP event), the safety-related solenoid valves are vented and the valve would fail in the prescribed fail- 7

function. Two valves in series are provided at each safety/non-safety inter-
face. These valves automatically close on a low-low surge tank level, pre-
venting the loss of the safety portion of the system and also limiting the
water that would be released into the building as a result of a failure in the
non-safety portion of the system. The isolation times of these valves are
adequate to preclude excessive drawdown of the surge tanks.

Inventory losses that result from a failure in the non-essential portions of
the CCWS can be compensated for by makeup water from the surge tanks. Makeup
water is normally provided to the surge tanks by the demineralized water makeup
system. However, when the demineralized water makeup system is unavailable, such as
during an accident, a backup make up-water line of seismic Category I design
can be provided by installing a spool piece to connect the SSWS to the CCWS
surge tank. Using the SSWS as a makeup water system provides a backup to the
intact division, since in the case of a major leak in one of the CCWS divi-
sions, the affected division is removed from service and the redundant
division is there to be used.

Leakage into or out of the CCWS is detected by level monitoring of both surge
tanks and sumps, as well as radiation monitoring. Safety-related instrumenta-
tion on the surge tank in each division alerts the operators to high, low, and
low-low levels in the tank that indicate system leakage. These monitors are
complemented by high-level alarms for the CCWS sumps and the CCWS heat
exchanger structure sumps. In addition, radiation monitors located downstream
of the CCWS pumps will identify any leakage of radioactive fluids into the
CCWS.

The CCWS cools the following reactor coolant pump (RCP) support systems: RCP
high pressure cooler; RCP seal coolers; ~~RCP lower bearing oil cooler~~; RCP
motor coolers; and RCP motor ~~upper bearing~~ oil coolers. These RCP support systems
are part of the non-essential portion of the CCWS. However, the supply and
return headers for the RCP support systems do not isolate on an safety
injection actuation signal (SIAS) and will be supplied with cooling water
following a small-break LOCA with an LOOP in accordance with the requirements
of 10 CFR 50.34(f)(1)(iii). Low- and high-flow alarms are provided for the
various RCP heat exchangers which use CCWS for cooling. These alarms alert

safe position. The motor-operated valves in the essential portions of the CCWS are powered from diesel-backed power sources and would be available subsequent to an LOOP.

The CCWS comprises two divisions that are spatially and electrically separated and share no components or systems. This design precludes any single event from affecting both systems. Each division is individually capable of providing the requisite heat removal capability to support a reactor shutdown and continued cooling following a design-basis accident (DBA). In addition, the design has redundant trains within each division ensuring that the failure of an individual component or train will not impair the functionality of the CCWS. Although the System 80+ design can be used at either single or multiple unit sites as described in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. Should a multi-unit site be proposed, the COL applicant would have 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 structures, systems, and components to perform their required safety functions. This was identified as COL Item 9.2.2-1 in the DSER. Upon further review the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the CCWS meets the requirements of GDC 5, and COL Action Item 9.2.2-1 is resolved. (8)

The redundant divisions also ensure that safety functions can be performed assuming a single active component failure coincident with the LOOP. Within each division, ABB-CE has designed component redundancy (e.g., two pumps, two heat exchangers) to ensure that a single failure would not typically compromise the heat removal function of a division.

This component redundancy is complemented by the motor-operated valves that isolate an individual division or individual trains within a division. These isolation valves protect the essential components from failures of the non-essential portions of the CCWS. The piping and instrumentation diagrams (P&IDs) clearly identify the class breaks between the essential and non-essential portions of the system and the valves that provide the isolation

function. Two valves in series are provided at each safety/non-safety inter-
face. These valves automatically close on a low-low surge tank level, pre-
venting the loss of the safety portion of the system and also limiting the
water that would be released into the building as a result of a failure in the
non-safety portion of the system. The isolation times of these valves are
adequate to preclude excessive drawdown of the surge tanks.

Inventory losses that result from a failure in the non-essential portions of
the CCWS can be compensated for by makeup water from the surge tanks. Makeup
water is normally provided to the surge tanks by the demineralized water
system. However, when the demineralized water system is unavailable, such as
during an accident, a backup make up-water line of seismic Category I design
can be provided by installing a spool piece to connect the SSWS to the CCWS
surge tank. Using the SSWS as a makeup water system provides a backup to the
intact division, since in the case of a major leak in one of the CCWS divi-
sions, the affected division is removed from service and the redundant
division is there to be used.

Leakage into or out of the CCWS is detected by level monitoring of both surge
tanks and sumps, as well as radiation monitoring. Safety-related instrumenta-
tion on the surge tank in each division alerts the operators to high, low, and
low-low levels in the tank that indicate system leakage. These monitors are
complemented by high-level alarms for the CCWS sumps and the CCWS heat
exchanger structure sumps. In addition, radiation monitors located downstream
of the CCWS pumps will identify any leakage of radioactive fluids into the
CCWS.

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The CCWS cools the following reactor coolant pump (RCP) support systems: RCP
high pressure cooler; RCP seal coolers; RCP lower bearing oil cooler; RCP
motor coolers; and RCP upper bearing oil cooler. These RCP support systems
are part of the non-essential portion of the CCWS. However, the supply and
return headers for the RCP support systems do not isolate on an safety
injection actuation signal (SIAS) and will be supplied with cooling water
following a small-break LOCA with an LOOP in accordance with the requirements
of 10 CFR 50.34(f)(1)(iii). Low- and high-flow alarms are provided for the
various RCP heat exchangers which use CCWS for cooling. These alarms alert

the control room of flow anomalies associated with the heat exchangers and ensure that timely notification of the cooling problem is provided to protect the RCP pumps.

Based on the above discussion, the system meets the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat to the UHS from structures, systems, and components important to safety.

Components of the CCWS can be fully tested during normal operation. The redundant trains of equipment within each division provides flexibility in the scheduling and conduct of inspections. In addition, tests to verify proper operation of individual CCWS components can be conducted using installed bypass and recirculation loops. These tests supplement the system level tests by verifying acceptable performance of each active component in the CCWS. The surveillance and testing requirements are discussed in Chapter 16 as part of the staff's review of CESSAR-DC Chapter 16. Therefore, GDC 45 is met.

The seismic design and isolation provisions between essential and non-essential portions of the CCWS have been reviewed and found acceptable. In addition, the staff verified that the CCWS will provide cooling to essential nuclear components during normal, off-normal, and accident conditions. Accordingly, the CCWS meets GDC 46.

In response to RAI Q410.111(b), ABB-CE added dual isolation valves in series where the essential and non-essential portions of the system meet. ABB-CE also identified several sections and tables of the CESSAR-DC that needed to be revised to add reference to the addition of the valves. However, ABB-CE did not provide any reference to these valves (CC-122, CC-123, CC-222 and CC-223) in CESSAR-DC Section 9.2.2.2.2.5, "Emergency Operation." This was DSER Open Item 9.2.2-1 in the DSER. By CESSAR-DC Amendment R, ABB-CE has provided the reference of these valves in Section 9.2.2.2.2.5, "Emergency Operation." The staff finds that the DSER Open Item 9.2.2-1 is resolved.

The design of the CCWS complies with GDC 45, 46, and 2 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The system design also meets the guidelines of Positions C.1 and C.2 of RG 1.29 with respect to seismic requirements for the safety related and applicable non-safety-related portions of the system. Further, the system design complies with GDC 44 and 4 with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures. Therefore, the staff concludes that the system design meets the applicable acceptance criteria of SRP Section 9.2.2.

Security Considerations

The staff considers that the CCWS is a vital system; therefore, as required by 10 CFR 73.55(c), access to all CCWS components, including pumps, piping, valves, heat exchangers, controls, power supplies, and other essential components and auxiliaries, should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desired, if access to each vital area is otherwise controlled.) The description in CESSAR-DC Section 9.2.2.3 (D) of the protected location of CCWS components did not address protection from sabotage. Vital designation of the CCWS was identified as DSER Open Item 9.2.2-2. By CESSAR-DC Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.2.1.1: "The CCWS is designated as a vital system and, therefore, will be protected from sabotage." By adding this statement ABB-CE has clarified that the CCWS is a vital system and resolves this item.

In a letter of February 28, 1992, ABB-CE proposed to add a new section to the CESSAR-DC (Section 9.2.2.1.4) which would state that the CCWS heat exchanger structure is a seismic Category I structure within the scope of the COL applicant and would require that applicant to provide a CCWS heat exchanger structure that is in the "vital protection area" and that will "withstand the effects" of a sabotage event. The staff interpreted that terminology to mean that access to equipment within the CCWS heat exchanger structure requires passage through both the protected area fence and an additional vital barrier. The proposed vital area designation is in accordance with the NRC Review Guideline 17 criterion that seismic Category I equipment be considered vital equipment; that designation is acceptable. However, the proposed description of the CCWS heat exchanger structure ventilation system and its fresh air intakes did not make reference to ventilation barrier guidance of RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," and was not sufficient for the staff to determine that a structure designed to these requirements will adequately protect the vital barrier. This was identified as COL Action Item 9.2.2-2. By CESSAR-DC Amendment Q, ABB-CE added the following statement to CESSAR-DC Section 9.2.2.1.4: "Ventilation barriers for the CCW Heat Exchanger Structure(s) Ventilation Systems are in accordance with the guidance provided in Regulatory Guide 5.65." By adding the this statement ABB-CE has clarified the plant design. Requiring the CCW Heat Exchanger Structure Ventilation System barriers to be designed in accordance with the guidance provided in RG 5.65 resolves this item and deletes the need for a COL Action Item.

9.2.3 Demineralized Water Makeup System

The DWMS supplies filtered demineralized water to the condensate storage system (CSS) and to other systems throughout the plant that require high-quality, non-safety-related, makeup water. The system consists of a demineralizer with cation, anion, and mixed-bed units, a vacuum degasifier, and a demineralized water storage tank. The DWMS does not perform any safety function or accident mitigation and its failure would not reduce the safety of the plant.

The staff evaluated the design and operational requirements of the system and finds that it includes all components and piping associated with the systems. The review has determined the adequacy of the proposed design criteria and design bases for the DWMS, regarding adequate supply of reactor coolant purity water during all conditions of plant operation.

The design of the DWMS is acceptable because it is in agreement with GDC 2 and 5 as recommended in SRP Section 9.2.3.

9.2.4 Potable and Sanitary Water Systems

The Potable and Sanitary Water Systems (PSWS) consist of a Potable Water System and a Sanitary Drainage system. It includes all components and piping from the filtered water source to all points of discharge to the sewage facilities. The portions of the PSWS that are within the Reactor Building, Nuclear Annex, Turbine Building, Radwaste Building, and Service Building are within the scope of the Certified Design. Those portions of the PSWS that are not within these buildings are not within the scope of the Certified Design.

The staff reviewed the design requirements for potable and sanitary water systems (PSWS) in accordance with SRP Section 9.2.4, "Potable and Sanitary Water Systems."

ABB-CE states in CESSAR-DC that the PSWS serve no safety functions and any malfunction of the systems will have no adverse impact on any safety-related system. Design provisions are provided to control the release of liquid effluent containing radioactive material from contaminating the PSWS by providing no interconnections with systems having the potential for containing radioactive materials (Interface 9.2.4-1). Additionally, where necessary, air gaps protect against the contamination of the potable water system with radioactive effluents (Interface 9.2.4-2). Designs meeting these requirements satisfy GDC 60. (8)

In the DSER, the staff stated that the design of the PSWS are site dependent and are, therefore, not described in detail in the CESSAR-DC. Specific PSWS designs will be reviewed as part of site specific applications referencing this design. This was identified as COL Action Item 9.2.4-1 in the DSER. By CESSAR-DC Amendments Q and T, ABB-CE has provided additional information regarding the PSWS and stated that the COL applicant will provide the information on those portions of the PSWS that are out of scope. ABB-CE has stated that the PSWS shall be designed to meet the requirements of GDC 60. Specifically, there shall be no interconnections between the potable and sanitary water systems and systems having the potential for containing radioactive materials. Additionally, the COL applicant shall ensure that the sewage treatment facility design complies with applicable state and local regulations. The CESSAR-DC has provided sufficient interface requirements to assure that plant specific designs for these systems will meet the requirements of GDC 60. Therefore, COL Item 9.2.4-1 is resolved.

9.2.5 Ultimate Heat Sink

The staff reviewed the design interface requirements for the UHS in accordance with SRP Section 9.2.5, "Ultimate Heat Sink."

The UHS is an out-of-scope item that will be reviewed for each site-specific application. Only a general discussion of the UHS appears in the CESSAR-DC. The UHS is the source of cooling to the SSWS which removes heat from the CCWS. The CCWS removes heat from essential and non-essential reactor auxiliary loads during all modes of plant operation.

The conceptual design of the UHS presented in the CESSAR-DC is a single, passive, independent cooling water pond and includes the SSWS intake and discharge. However, ABB-CE notes that site-specific conditions may necessitate the use of two ponds to satisfy RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The pond has ~~redundant~~ water makeup pumps to maintain water level. Water chemistry is maintained by a site-specific water treatment system, and salinity buildup in the pond is limited by blowdown. The UHS will be designed to operate for the required nominal 30 days following a postulated LOCA without requiring any makeup water to the source, and without requiring any blowdown from the pond salinity control system.

The UHS shall meet seismic Category I requirements. In addition, the function of the UHS will not be lost during or after natural phenomena, including a safe shutdown earthquake, tornado, flood, or drought. Accordingly, the UHS satisfies RG 1.29, Position C.1, and RG 1.27, Positions C.2 and C.3.

As presented in the CESSAR-DC, the design of the UHS indicates that there are no shared systems or components, in accordance with GDC 5. In addition, the design of the UHS will ensure the continued operability of the system assuming a single failure of a manmade structure.

The UHS shall provide an SSWS inlet temperature that does not exceed the maximum allowable temperature required for removing heat from the CCWS heat exchangers during a DBA concurrent with an LOOP. This heat removal capacity includes heat loads anticipated from the start of the accident through the

Because the TBCWS is not safety related and does not share boundaries with a safety system, the remaining requirements (GDC 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply. Therefore, the staff concludes that the TBCWS meets the applicable requirements of SRP Section 9.2.2

9.2.9 Chilled Water System

The staff reviewed the design of the chilled water systems (CWSs) in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems."

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 chilled water system (ECWS) that provides safety-related heating, ventilation, and air conditioning (HVAC) cooling loads, and a normal chilled water system (NCWS) that provides non-safety-related HVAC cooling loads.

9.2.9.1 Essential Chilled Water System

The ECWS consists of two equally sized divisions. Each division is sized to provide 100 percent of the cooling capacity required to meet system demands during normal and accident conditions. Each division is supplied electrical power from independent Class 1E power sources and cooling water from the respective CCWS division ~~trains.~~ The ECWS provides chilled water to the safety-related HVAC cooling loads in the control room, ~~computer room,~~ electrical rooms, mechanical rooms, subsphere pump rooms, and penetration room.

The ECWS is located in a flood- and tornado-missile protected seismic Category I structure. The ECWS is designed in accordance with seismic Category I and Class 1E requirements. The ECWS is protected from pipe breaks, pipe whip, tornado missiles, jet impingement, and severe environmental conditions. ABB-CE did not, however, indicate if the system's design considers potential water hammer concerns. This was Open Item 9.2.9.1-1 in the DSER. By CESSAR-DC Amendment Q, ABB-CE has stated that the ECWS is designed to minimize the consequences of potential water hammer. Therefore, the DSER Open Item 9.2.9.1-1 is closed.

Additionally, in CESSAR-DC Section 9.2.9.1(B), the reference to "safety-related portions" of the ECWS was inconsistent with the reference to the ECWS as a "safety-related system" in Section 9.2.9.2. Neither CESSAR-DC Figure 9.2.9-1 nor the flow diagrams provided in response to RAI Q410.113 clarified which portions of the ECWS were safety related. This was DSER Open Item 9.2.9.1-2 in the DSER. By CESSAR-DC Amendment Q, ABB-CE identified safety-related components and non-safety- related components of the ECWS separately in Figure 9.2.9-1. Therefore, DSER Open Item 9.2.9.1-2 is closed.

Based on the above discussion, the design presented in the CESSAR-DC satisfies GDC 2, with respect to its seismic requirements, by virtue of meeting the guidance of RG 1.29 Position C.1, regarding safety-related portions of the system, and GDC 4 regarding environmental and dynamic effects.

~~In CESSAR-DC Section 9.2.9.2.1(E),~~ ABB-CE indicates that the ECWS and the NCWS are indirectly connected through a heat exchanger ~~and pump.~~ As shown in CESSAR-DC Table 3.2-1, the ECWS heat exchanger is designed to seismic Category I Safety Class 3. The ECWS heat exchanger is designed to allow the NCWS to serve all of the ECWS during periods of normal operation without directly connecting the water pathways. Therefore, the integrity of the safety-related ECWS would not be degraded by postulated failures in the NCWS.

The System 80+ design can be used at either single- or multiple-unit sites; however, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. Should a multi-unit site be proposed, the COL applicant would have 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 structures, systems, and components to perform their required safety functions. This was COL Action Item 9.2.9.1-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear units. Therefore, the ECWS meets the requirements of GDC 5.

a chemical addition tank, piping, valves, controls, and instrumentation

Each 100-percent-capacity ECWS division consists of a ~~chilled water refrigeration unit~~ chiller, a heat exchanger, ~~a circulating~~ two chilled water pumps, ~~a compression~~ an expansion tank, ~~control valves, instrumentation, and piping~~ a chemical addition tank, piping, valves, controls, and instrumentation. ~~Additionally, there is a~~ The ECWS heat exchanger and ~~heat exchanger~~ associated chilled water pump ~~that~~ allow~~s~~ the NCWS to supply 100 percent of the normal ECWS loads without directly connecting the water pathways. In CESSAR-DC Section 9.2.9.2.1(E), ABB-CE states that the ~~heat exchanger~~ pump that series the essential chilled water heat exchanger can serve as a backup ~~ECWS pump~~ for the pump that series the essential chiller. However, the flow diagram in CESSAR-DC Figure 9.2.9-1 was not sufficient to show how the cross-connect valve between the pump discharge lines would prevent backflow through the secured pump. ABB-CE was asked to provide P&IDs for both the ECWS and NCWS. This was DSER Open Item 9.2.9.1-3 in the DSER. By CESSAR-DC Amendment S, ABB-CE has provided the updated Figure 9.2.9-1 which shows both ECWS and NCWS. Therefore, DSER Open Item 9.2.9.1-3 is closed and the ECWS system meets the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat from structures, systems, and components important to safety to the UHS.

ABB-CE indicates that the ~~CSS~~ ECWS provides access necessary to support inservice inspections and functional testing of safety-related components and equipment. Therefore, the ~~CSS~~ ECWS satisfies GDC 45 and 46.

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 from postulated piping failures. The design also complies with GDC 5, 44, 45, and 46 with respect to shared systems, cooling water requirements, and inservice inspection and testing requirements. Therefore, the staff concludes that the system design meets the applicable acceptance criteria of SRP Section 9.2.2.

9.2.9.2 Normal Chilled Water System

The normal chilled water system (NCWS) consists of two equally sized divisions. Each division is sized to provide 100 percent of the cooling capacity required to meet system demands during normal conditions. The NCWS is a non-safety-related system. However, the containment cooling systems serviced by this system are designed to operate during an LOOP. The power supply to the NCWS pumps and chiller units is automatically transferred ~~(except during~~

~~accident conditions)~~ to the ~~standby~~ alternate ~~ac~~ power source when normal power is not available. Each division is supplied cooling water from the respective CCWS division ~~trains~~. The NCWS provides chilled water to the non-safety-related HVAC cooling loads in the ~~containment, control element drive mechanism (CEDM), high purge, fuel building~~, nuclear annex, reactor building ~~break room, change rooms, conference area, technical support,~~ and radwaste building.

The NCWS system is not safety related because it is not required to ensure: (1) the integrity of the RCS pressure boundary is maintained; (2) the capability to achieve and maintain safe shutdown; and (3) the ability 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. Each division of the NCWS enters the primary containment through two penetrations: one for the supply line; and the other for the return line. ~~The supply line penetration has one motor-operated isolation valve outside the containment and a check (isolation) valve inside the containment. The return line penetration has two motor-operated isolation valves, one inside and one outside the containment, and one check valve inside the containment.~~ Isolation valves and piping for the primary containment penetrations are safety related and are designed to seismic Category I Safety Class 2 and 10 CFR Part 50 (Appendix B) standards. (10)

The rest of the system is not safety related, as stated above. However, the non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety, are designed in accordance with seismic Category II criteria ~~to ensure their integrity under seismic loadings resulting from a safe shutdown earthquake.~~ On this basis, the staff finds that the design of the NCWS system meets Positions C.1 and C.2 of RG 1.29, as addressed by the SRP Section 9.2.2 acceptance criterion with respect to the seismic requirements for the safety-related and non-safety-related portions of the system.

By virtue of their location in seismic Category I, flood- and tornado-missile-protected structures, the safety-related portions of the system are protected against damage from natural phenomena. Further, all safety-related systems are protected against flooding that may result in the event of system failure, (including failures that could cause flooding) should not lead to the failure of any safety-related structure. This was Interface 9.2.10-1 in the DSER. By CESSAR-DC amendments up to and including Amendment L, ABB-CE revised CESSAR-DC to state that the TBSWS is located in a building that does not contain any safety-related components. Therefore, this Interface Item 9.2.10-1 is no longer required.

The system will meet GDC 2 by meeting the requirements of RG 1.29, Position C.2 for assuring that the non-safety-related portions of the system withstand the effects of earthquakes without affecting adjacent safety-related systems.

Because the TBSWS is not safety related and does not share boundaries with a safety system, the remaining requirements (GDC 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply. Therefore, the staff concludes that the TBSWS meets the applicable requirements of SRP Section 9.2.2

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The staff reviewed the compressed air systems in accordance with SRP Section 9.3.1. Conformance with the acceptance criteria formed the basis for the evaluation of the compressed air systems with respect to the applicable regulations, specifically: GDC 1 for quality standards, GDC 2 for earthquake resistance, and GDC 5 for the capability of shared systems and components, important to safety, to perform required safety functions.

The staff based its review on ABB-CE's response to RAI Q410.114. In the response, ABB-CE submitted extensive revisions to the CESSAR-DC. This was identified as Confirmatory Item 9.3.1-1 in the DSER. ABB-CE incorporated these revisions in Section 9.3.1 of Amendment J to the CESSAR-DC. The staff finds this acceptable to resolve Confirmatory Item 9.3.1-1.

The compressed air systems ~~comprises~~ consist of the instrument air, station air, and breathing air systems. The instrument air system supplies clean, oil-free, dried air to all air-operated instrumentation and valves. The station air system supplies compressed air for air-operated tools, and miscellaneous equipment, and for various maintenance purposes. The breathing air system supplies clean, oil-free, low-pressure air to various locations in the plant to protect employees against contamination while they performing certain maintenance and cleaning operations.

The instrument air system consists of four parallel trains of 100-percent-capacity air compressors of oil-free, water-cooled design; an air receiver; and an instrument air dryer connected in series. Each compressor has an intake filter/silencer rated to remove all particles that exceed 5 microns (0.2 mils). Downstream of each air compressor, the instrument air flows into an instrument air receiver that has adequate reserve capacity to allow the standby compressors to be started following a compressor trip. Downstream of the air receivers, the instrument air passes through an instrument air dryer. Each air-dryer is equipped with a coalescing prefilter, an air dryer assembly, and an afterfilter capable of drying the compressed air to a dewpoint of -40 °~~C~~F ~~(-40 °F)~~ at line pressure and filtering the air of particulates that exceed 1 micron (0.04 mils). Therefore, the design presented in the CESSAR-DC complies with the guidance of ANSI MC 11.1-1976 (ISA S7.3) which requires a clean, dry, oil-free air supply to safety-related components.

Instrument air lines penetrating the containment are equipped with electric-operated isolation valves (outside the containment) and check valves (inside the containment). The compressors are powered from non-safety-related buses, but they can be manually aligned to the non-Class 1E alternate ac (AAC) source standby power supply during an LOOP.

The station air system consists of two oil-free, 100-percent-capacity, station air compressors, each consisting of an intercooler, aftercooler, and moisture separators. Downstream of the compressors, the air flows to air receivers and is then dried by one of two redundant station air dryers before it is distributed throughout the plant via station air headers.

The breathing air system consists of two, oil-free, 100-percent-capacity breathing air compressors, each consisting of an intercooler, aftercooler, and provisions that may be necessary to allow this capability, such as shielding of instrument transmitters and logic cabinets from radio frequency interference (RFI), fiber-optic cabling, and radio repeaters within buildings, and did not provide reasons for deviating from the EPRI ALWR URD. NRC Information Notice 83-83 stated: "As newer plants are built that use more solid state equipment . . . more cases of RFI by portable radio transmitters are likely to result . . . . If plant operations make the use of portable radio transmitters near RFI-sensitive equipment either necessary or likely in an emergency, then administrative prohibitions are not adequate and the licensee should consider hardware fixes." This was identified as DSER Open Item 9.5.2-3.

Subsequently, in Amendment J to CESSAR-DC, ABB-CE added the following statement Section 13.6: "The security communications subsystem shall meet the following requirements: 1. Each on-site security officer, watchman, or armed response individual shall be provided with continuous communications with an individual in each continuously manned alarm station. This may be accomplished by using multi-frequency radio or microwave transmitted two-way voice communications." Requiring continuous wireless communication between security officers, will ensure adequate communication for the security organization. This additional information is acceptable to the staff. On this basis, DSER Open Item 9.5.2-3 is resolved.

9.5.3 Lighting System~~s~~

The normal lighting system will supply normal illumination under all plant operating, maintenance, and test conditions. CESSAR-DC Table 9.5.3-1 summarizes typical illuminance ranges for normal lighting. The lighting fixtures are designed and located so that plant personnel can maintain and replace lights effectively and safely. ABB-CE indicates that the circuits to the individual lighting fixtures will be staggered as much as possible and that separate electrical circuits will feed these staggered circuits to ensure that some lighting is retained in the event of a circuit failure. The failure or unavailability of a single lighting transformer will not affect the ability of the system to operate normally. The normal lighting system will be part of be accomplished by two systems: conventional AC fixtures fed from Class 1E power sources and Class 1E DC self contained, battery operated lighting units.

the plant's permanent non-safety system. Therefore, the normal lighting system will have power as long as power from an offsite power source or a standby non-safety source (combustion turbine generator (CTG)) is available.

ABB-CE indicated that the emergency lighting system for System 80+ design will ~~be Class 1E and will be powered by an EDG~~. This emergency lighting will be located in vital areas throughout the plant. These areas are determined by performing hazard analyses and establishing plant emergency procedures. ABB-CE indicates that the operator will need to gain access to several vital areas, i.e., the main control room (MCR), the technical support center, the operations support center, the remote shutdown panel room, the sample room, the hydrogen recombiner rooms, and stairwells and passageways. ABB-CE states that as it completes hazard analyses and plant emergency procedures, it is designating other areas as vital, such as the EDG rooms, the steam-driven emergency feedwater pump rooms, and the pathways from the control room to these rooms.

ADD INSERT E

~~The security lighting system will be considered part of the permanent non-safety system and will be fed from an uninterruptible power supply connected to a non-safety-grade battery. The security lighting system will remain energized as long as power from an offsite power source, a standby non-safety source (CTG), or a non-safety-grade battery is available.~~

ABB-CE submitted information on the design of the normal, emergency, self-contained dc lighting units and security lighting to demonstrate that the lighting in the normal and vital plant areas as well as passageways to and from these areas are adequate. After installation, each lighting system will be inspected, checked, and tested to verify that it is operable and provides proper coverage.

In the DSER, the staff requested ABB-CE to address the following staff concerns and EPRI guidelines:

1. The staff will verify that the completion of the hazard analyses and plant emergency procedures are included as commitments and that appropriate inspections, tests, and/or analyses are included as part of ITAAC to verify implementation of their design commitments.

2. State what method will be used to distinguish between the normal, emergency, and security lighting cables and circuits to ensure they are physically identified and separated.

3. Confirm that the Class 1E distribution system supplies at least one of the circuits supplying lighting fixtures for the normal lighting system in safety-related areas (other than main control room) and in access routes to these areas. The other lighting circuit can be supplied from a non-Class 1E electrical division backed-up by the CTG.

4. Integrate the emergency lighting system in the main control room with the normal lighting system and design it so that alternate emergency lighting fixtures are fed from separate safety divisions.

5. Design the emergency lighting installations that serve the main control room and those other areas of the plant where safe shutdown operation may be performed so that these installations will continue to function during and after a DBE.

6. Confirm which part of the emergency lighting system will not be qualified as Class 1E.

The staff stated in the DSER that it would verify that the above aspects are included in the design commitments. This was designated as DSER Open Item 9.5.3-1.

In Amendment T, ABB-CE supplemented the information on lighting systems as follows:

ABB-CE ~~has completed the hazard analyses and plant emergency procedures and~~ has designated that the following areas of the plant will be included as vital areas; the main 14 control room, the technical support center, the operations support center, the remote shutdown panel room, the sample room, the hydrogen recombiner rooms, electrical system areas, main steam valve houses, chemistry labs, the EDG rooms, the stairway which provides access from the control room to the remote shutdown panel room, and other areas where operator access is required. ABB-CE indicated that associated Class 1E emergency lighting will be located in vital areas of the plant. The associated Class 1E emergency lighting system is used to provide acceptable levels of illumination throughout the station and particularly in vital areas where emergency operations are performed upon loss of the normal lighting system. The associated Class 1E emergency lighting system provides a minimum illumination level of 10 foot-candles in areas of the plant where emergency operations are performed. For other areas of the plant covered by the emergency lighting system, a minimum illumination level of 2 foot candles is provided. This addresses Item (1) and is acceptable. The adequacy and acceptability of the System 80+ design description and ITAAC are included in Chapter 14 of this report.

ABB-CE indicated that the criteria for the physical identification of lighting cables and circuits are consistent with the criteria for physical identification and separation of Class 1E and non-Class 1E cables and circuits as described in IEEE-384 and RG 1.75, which are part of CESSAR-DC, Chapter 8, Electrical Power Systems. This addressed item (2) and is acceptable.

~~ABB-CE indicated that the circuits to the individual lighting fixtures in safety-related areas and access routes to these areas are staggered as much as possible, with the staggered circuits fed from separate electrical divisions. One of the circuits feeding the lighting fixtures are supplied from the Class 1E distribution system and is backed up by the EDGs. The other lighting circuit is supplied from a separate non-Class 1E electrical division and is backed up by the CTG. In addition,~~ ABB-CE indicated that all lighting fixtures and other components of the lighting system located in normally occupied areas or in areas containing safety equipment are supported so as to enhance the earthquake survivability of these components and to ensure that they do not present a personnel or equipment hazard when subjected to a seismic loading of a DBE. This addresses item (3) and is acceptable.

ABB-CE indicated that the emergency lighting system in the main control room will be integrated with the normal lighting system. The emergency lighting in the MCR will be configured so that at least two circuits of lighting fixtures are powered from different Class 1E divisions to ensure lighting is retained in the event of a circuit failure. This addresses item (4) and is acceptable subject to incorporation into the next CESSAR-DC amendment (Amendment V). This is part of FSER Confirmatory Item 1.1-1.

ABB-CE indicated that the associated Class 1E emergency lighting system in the main control room will maintain minimum illumination levels in the main control room during all operating and emergency conditions, including a station blackout (SBO). The Class 1E and associated Class 1E emergency lighting installations which serve the main control room and other areas of the plant where safe shutdown operations may be performed are designed to remain functional during and after a DBE. This addresses item (5) and is acceptable.

1E

ABB-CE indicated that the Emergency lighting is composed of two systems, (1) conventional ac fixtures fed from Class IE ac power sources which, excluding the fixtures, are qualified as associated Class 1E circuits and, (2) dc self contained battery-operated lighting units which are qualified as Class 1E circuits. This addresses item (6) and is acceptable.

ABB-CE indicated that the lighting system for the System 80+ is designed to provide illumination throughout the plant and plant site. In accordance with SRP Section 9.5.3, lighting levels and illuminance ranges for the System 80+ design comply with the Illuminating Engineering Society (IES) Lighting Handbooks recommended intensities.

ADD INSERT G

ABB-CE indicated that ~~incandescent lighting is used in the Containment Building while incandescent, fluorescent and high intensity discharge lighting is provided in the remainder of the plant and plant site.~~ Fluorescent luminaries are normally used in plant stairs and stairwells; around switchgear, motor control centers, and instrumentation racks to supplement

; and

high intensity discharge (HID) luminaries. ~~This is done in order to provide partial illumination in areas where HID luminaries are in the process of starting or re-starting following a momentary loss of power.~~

ABB-CE indicated that the Class 1E dc self contained battery-operated lighting units ~~will illuminate stairways, exit routes, major control areas, and other dc areas where operator action is required. Each Class 1E unit is~~ are designed for eight hours of continuous operation following loss of normal power. Each Class 1E unit will have ~~sealed-beam lamps and~~ a self-contained battery pack unit containing a rechargeable battery with a minimum eight-hour capacity. The Class 1E lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located. The loading of these Class 1E lighting units will not be greater than 80 percent of the rated capacity with additional derating for temperature variations, where appropriate. The bulbs will be positioned so that adequate illumination is provided and is not obstructed by plant equipment and components. The Class 1E units will also contain a time delay so that the lights turn off on the resumption of power only after there is adequate time for normal lighting to restart.

ABB-CE indicated that additional non-Class 1E dc self-contained battery-operated lighting units will be installed throughout the plant to provide emergency lighting for personnel safety in accordance with the applicable sections of the National Electric Code and the Life Safety Code of the National Fire Protection Association.

Based on the above, the staff concludes that the lighting system for the CE System 80+ is in accordance with SRP Section 9.5.3 and the (IES) Lighting Handbooks and is acceptable. On this basis, DSER Open Item 9.5.3-1 is resolved.

Security Considerations

In CESSAR-DC Section 9.5.3.1, ABB-CE stated that the security lighting system will provide illumination required to monitor the isolation zones and ~~the protected area under normal conditions as well as upon loss of all ac~~ all outdoor areas within the plant protected perimeter and will comply with the intent of NUREG/CR-1327. By means of Amendment E, ABB-CE stated in CESSAR-DC Section 9.5.3.2.2 that the security lighting system is part of the permanent non-safety systems loads and is fed from an uninterruptible power supply (UPS) connected to a non-safety battery. In a December 17, 1991, response to RAI Q500.20, ABB-CE proposed to delete the reference to an UPS and change Section 9.5.3.2.2 to read instead: "The security lighting system is considered part of the permanent non-safety systems and is fed from the AAC source (combustion turbine), which is located in a secure vital area for protection." However, as described in CESSAR-DC Section 8.3.1.1.5.1, the AAC could take 10 minutes to start (from the onset of an LOOP event), and additional time for load sequencing before security lighting would be restored. The staff concluded that the proposed change is inconsistent with CESSAR-DC Section 9.5.3.1 and did not conform with the URD requirement for uninterruptible power for those portions of the security lighting that are essential to plant protection following interruption of normal power. Inconsistency between CESSAR-DC Section 9.5.3.1 and the changes proposed for CESSAR-DC Section 9.5.3.2.2, and ABB-CE's reason for deviating from the URD requirements for uninterruptible power for those portions of the security lighting that are essential to plant protection, was identified as DSER Open Item 9.5.3-2. By CESSAR-DC Amendment J, ABB-CE stated in CESSAR-DC Section 8.3.1.1.5.1, that the AAC is designed to start automatically within two minutes from the onset of an LOOP event. As stated in CESSAR-DC Section 9.5.3.2.2, the security lighting is part of the permanent non-safety systems which is fed from the AAC. Selected portions of the security lighting system essential to maintaining adequate plant protection are ~~power~~ Powered from an uninterruptible power supply. Requiring uninterruptible power for those portions of the security lighting that are essential to plant protection is consistent with the URD requirements. This additional information has also removed the inconsistency between CESSAR-DC Section 9.5.3.1 and the changes proposed for CESSAR-DC Section 9.5.3.2.2. The staff considers DSER Open Item 9.5.3-2 resolved.

In CESSAR-DC Section 9.5.3.2.2, ABB-CE stated that the security lighting system will provide a minimum illumination of 0.2 foot-candle at ground level. In CESSAR-DC Table 9.5.3-1, ABB-CE lists 2 to 5 foot-candles as typical illuminance ranges for normal exterior area lighting. Those illumination levels would give a range of ratios of typical to minimum illumination of 10:1

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10 STEAM AND POWER CONVERSION SYSTEM

10.1 Introduction

The steam and power conversion system converts the heat energy generated by the nuclear reactor into electric power. The heat energy produces steam in two steam generators, capable of driving a turbine generator unit.

The steam and power conversion system utilizes a condensing cycle with regenerative feedwater heating. The turbine exhaust steam is condensed in a surface type condenser. The reference design includes a circulating water system utilizing cooling towers. Alternate designs, as appropriate for each site, will be discussed in each site-specific application. The condensate from the steam is returned to the steam generators by means of the condensate and feedwater system (CFS).

A turbine bypass system (TBS), consisting of eight turbine bypass valves and associated piping, relieves a combined capacity of 55 percent of total full power steam flow at 6895 kPa (1000 psia) to the condenser during startup, shutdown, load shedding and transient conditions on the turbine, reactor, or both.

Each line combines to a common equalization header. Downstream of the header 4 steam lines are routed to the turbine trip.

Each steam generator has two main steam supply lines to route the steam to the turbine generator unit. Each of the four lines has a flow-measuring device, five spring-loaded main steam code safety relief valves, a main steam isolation valve (MSIV), a power-operated atmospheric dump valve (ADV), the main steam stop valve, and a control valve (located just upstream of the high-pressure turbine). The main steam code safety relief valves provide overpressure protection for the shell side of the steam generators and the main steam piping up to the inlet of the main steam stop valves. The ADVs are used (1) as a heat sink for steam generator cooldown when the MSIVs are closed or the main condenser is not available, (2) to hold the plant at hot standby or to perform a plant cooldown during a loss-of-offsite power or station blackout (SBO) event, and (3) to provide for a controlled cooldown in the event of a main steam line break or steam generator tube rupture.

The safety-related portions of the steam and power conversion system consist of (1) the emergency feedwater system (EFWS), including the main feedwater isolation valves (MSIVs) and associated piping to the steam generators, (2) the MSIVs, including the associated piping to the steam generators, (3) the ADVs, (4) the main steam code safety relief valves, and (5) the steam supply to the EFWS.

is provided in Figure 10.1-1.

The steam and power conversion system description is located in Section 10.1 of the CESSAR-DC. The system design and performance characteristics are identified in Table 10.1 1 of the CESSAR-DC. The reference heat balance diagram, based on the use of a cooling tower, and the main steam and feedwater system flow diagram are identified in CESSAR-DC Figures 10.1-1 and 10.1-2, 10. respectively. and 10.4.7-1,

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10.2 Turbine Generator

The turbine generator system (TGS) converts the energy in steam from the nuclear steam supply into electrical energy. The TGS includes all normally provided components and equipment, including the turbine main steam stop and control valves, and the intercept stop and control valves. The turbine generator consists of a double-flow, high-pressure turbine and three double-flow, low-pressure turbines driving a direct-coupled generator. The system is designed so that the single failure of a main stop, main control, intercept stop, or intercept control valve does not disable the turbine overspeed trip function. The TGS functions, under normal, upset, emergency, and faulted conditions, are monitored and controlled automatically by the turbine control system, which includes the redundant mechanical and electrical trip devices to prevent excessive overspeed of the TGS. The TGS description is located in Section 10.2 of the CESSAR-DC. The design parameters are identified in CESSAR-DC Tables 10.2.2-1 and 10.2.3-1. CESSAR-DC Figure 10.1-1 provides the reference flow diagram. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the TGS.

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INSERT 10.2 A

from the sensing circuits, from the speed control unit, or from devices detecting the state of plant components, (3) Computing functions to separate flow reference signals for the valve set,

INSERT 10.2 B

In addition the following tests are performed during normal operation, (1) the main stop, control, and intercept stop and control valve are stroked from the control room during normal operation once every two weeks, and (2) a signal to allow the extraction line power assisted check valves valves to partially close will be simulated from the main control room once per month.

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The TGS has a turbine control and overspeed protection system to control all normal and abnormal operating conditions and to ensure that a full-load rejection will not cause the turbine to overspeed beyond acceptable limits.

The electro-hydraulic control (EHC) system load control unit consists of two redundant turbine speed controllers (basic controllers). The turbine speed controllers, including the valve position controllers, use a 1-out-of-2 scheme of redundancy. There is an automatic switchover between the controllers in case of a disturbance on one controller. One automatic controller (load, frequency, pressure, limiters, etc.) provides a set value to the basic controller which provides positioning signals to the main control valves. The unit master interface is made via the automatic controller. The load control unit provides (1) sensing functions to detect and generate signals proportional to unit loading, (2) limiting functions to constrain flow reference signals in response to signals for the valve sets, and (4) logic function to ensure proper permissives have been satisfied prior to changes in mode of operation and provides switching signals to EHC system devices.

Insert 10.2A

There are four lines of defense against overspeed during all modes of operation including the 2 turbine speed controllers, a mechanical overspeed trip at 110 percent of rated speed, and electronic overspeed protection in a 2-out-of-3 logic scheme at 112 percent of rated speed. If the generator load is lost while the unit is in operation, an accelerator limiter (built into the EHC) senses the sudden load rejection and closes the control and intercept valves at the maximum rate. The entrained steam in the turbine casing, between the valves and the turbine, and in the crossover and extraction piping expands in less than 2 seconds. The expected overspeed is less than 10 percent (at full load), and the intercept valves will reopen when the actual speed is below the set value. If the speed control on the control and intercept valves malfunctions when generator load is lost, the turbine will accelerate until the mechanical overspeed trip activates to trip these valves, including their actuators, and the turbine will coast down to zero speed. If the turbine continues to accelerate, the electronic overspeed protection, in 3 2-out-of-2 logic, actuates the tripping device which causes the common safety system to be depressurized and all stop and control valves close very rapidly.

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The turbine overspeed protection includes redundant mechanical and electronic overspeed protection. The mechanical overspeed protection device closes all stop and control valves by depressurizing both the hydraulic emergency system and, via an interface relay, the common hydraulic safety system. The mechanical overspeed trip device is set to activate at 100 percent of rated speed. The mechanical overspeed trip device is fed from the lube oil system and the interface between the lube oil system and the turbine safety system is made via a separating relay. The electronic overspeed protection uses the three binary signals from the speed coordinating units. However, these signals go to the 2-out-of-3 tripping device in the common safety system and not in the EHC. The electronic overspeed protection, which is set at 112 percent of rated speed, is provided as a backup to the mechanical overspeed trip device. Additionally, these are two redundant reverse power relays for tripping the generator breaker to prevent overspeed after turbine trip and to prevent overheating of the last two stages of LP turbine blades. Each of the reverse power relays has two strategies, (1) reverse power for more than 1 second and depressurize turbine safety system, and (2) reverse power for more than 15 seconds. The turbine speed control system protection devices are listed in CESSAR-DC Table 10.2.2-1.

coordinating

The EHC system incorporates several provisions to provide basic turbine control functions including automatic control of turbine speed and acceleration through the entire speed range, and load and loading rate from auxiliary to full load. The EHC system provides fluid for turbine controls at 4000 kPa (580 psig). The turbine speed is measured by three independent speed modules including sensors and conditioning devices. The EHC is a microprocessor based controller. The video-operated control room control panel keyboard or push button determine the increase or decrease inputs. The runbacks are determined by logic of speed control, load reference signal exceeding a preset load limit, loss of generator status coolant, process control system signal and partial loss of load.

To prevent excessive decrease in steam throttle pressure, a main steam pressure limiter circuit is provided to close the controlling valve set when the throttle pressure falls below a set point. The turbine and its control valves are designed to pass the rated steam flow at throttle pressures vacuum pump air discharge is continuously monitored for radiation to detect steam generator primary-to-secondary tube leaks as discussed in Section 10.4.2 of this report.

The turbine generator is located in the turbine building and is separated from all portions of safety-related systems; no safety-related structures, systems, or components are located in the turbine building. Therefore, in the event of a failure of a high- or moderate-energy line, safety-related components will not be affected. The turbine orientation and the design of the safety-related structures provide protection against turbine missiles.

stop and control the

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The program for inservice inspection of the main steam and intercept valves includes, at approximately 3-1/3-year intervals, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule (required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code)) for reactor components, (1) at least one main steam stop valve, one main steam control valve, one intercept stop valve, and one intercept control valve be dismantled, and (2) visual and surface examinations of the valve seats, disks, and stems be conducted. In addition, (1) the main stop, control, and intercept stop and control valve are exercised at least once a week by closing each valve and observing by the valve position indicator that the indicator moves smoothly to a fully closed position, and monthly observing actual valve motion, and (2) the extraction steam check valves are tested weekly with the turbine on line and loaded.

and control

Insert 10.2B

Refer to CESSAR Pg 10.2-18

The design of the TGS is acceptable and meets the requirements of General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Bases," with respect to the protection of structures, systems, and components important to safety from the effects of turbine missiles. The system meets the requirements by providing a turbine overspeed protection system to control the turbine action under all operating conditions, which ensures that a full-load turbine reject will not cause the turbine to overspeed beyond acceptable

ABB-CE has met the requirements of GDC 4 (10 CFR Part 50) with respect to the use of materials with acceptable fracture toughness and elevated temperature properties, adequate design, and the requirements for preservice and inservice inspections. ABB-CE has described its program for ensuring the integrity of low-pressure turbine discs by the use of suitable materials of adequate fracture toughness, conservative design practices, and preservice and inservice inspections. These provisions provide reasonable assurance that the probability of failure with missile generation is low during normal operation and transients up to design overspeed.

10.3 Main Steam Supply System

The main steam supply system (MSSS) transports steam from the secondary side of the NSSS to the turbine generator; other safety-related or non-safety-related MSSS auxiliaries dissipate heat during cooldown, following a turbine and/or reactor trip and during main condenser unavail-ability, isolate steam generators, as necessary (e.g., containment isolation, post-loss-of-coolant accident (LOCA)), and provide overpressure protection for the shell side of steam generators and main steam piping. The safety-related portion of the system includes a portion between the steam generators down to and including the MSIVs. In accordance with CESSAR-DC Table 3.2-1, the essential portions of the MSSS are designed to Safety Class 2, seismic Category I, and are subject to the quality assurance requirements of 10 CFR Part 50 (Appendix B). The system description, design parameters, and reference flow diagram are given in CESSAR-DC Section 10.3.2, Tables 10.3-2 and 10.3.2-1, and Figures 10.1-2 and 10.3.2-1, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the MSSS.

The MSSS conveys high-pressure steam generated in two steam generators through two lines for each steam generator to a high-pressure (HP) turbine. Extraction steam from the HP turbine heats the feedwater in the HP feedwater heaters. Each line is equipped with (1) five ASME Code spring-loaded main steam safety valves (MSSVs) to protect against overpressurization of the shell side of the steam generators and the main steamline piping; (2) an atmospheric steam dump valve which is located between the safety valves and the MSIV; and (3) an MSIV and a bypass valve to establish positive isolation against both

4. The MSSVs and ADVs are arranged such that any condensate in the line between these valves and the main steam lines drain back to the main steam lines.

5. The MSSS is designed to accommodate steam hammer dynamic loads and relief valve discharge loads resulting from the rapid closure of system valves and safety/relief valve operation without compromising safety functions.

Also, the valves are periodically inservice tested for freedom of movement during plant operation in accordance with ASME Code Section XI, "Rules of Inservice Inspection of Nuclear Power Plant Components," Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants." However, in the DSER, the staff stated that ABB-CE had not mentioned any activity or program regarding personnel awareness in relation to potential occurrence of steam hammer dynamics. ABB-CE needed to clearly state that such a program was an activity for the COL applicant and provide guidance for developing plant operating and maintenance procedures which protect against a potential

10.3-8

occurrence of steam hammer. This was identified as ~~DSER Open Item~~ [redacted]nd is COL Action Item 10.3-1. In an amendment to the CESSAR-DC, ABB-CE stated that the COL applicant shall provide plant operating and maintenance procedures which ~~provide actions to be taken by~~ the plant personnel on the potential for the occurrence of the steam hammer and water entrainment, and the means to minimize such occurrences. On this basis, DSER Open Item 10.3-2 is resolved.

educate

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must 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 structures, systems, and components to perform their required safety functions. This was identified as C[redacted] in the DSER. Upon further review, the staff has determined that the design described in the

CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the MSSS meets the requirements of GDC 5, and ~~COL Action Item 10.3-2 is no longer necessary~~.

The MSSS is capable of providing heat sink capacity for the nuclear reactor, pressure relief for the shell side of the steam generator and the main steam lines up to the MSIVs, and supplying steam to the steam-driven safety-related pumps necessary for safe shutdown. Therefore, ABB-CE has met the requirements of GDC 34, "Residual Heat Removal," with respect to the system function of transferring residual and sensible heat from the reactor system. The modulating ADV (one for each main steamline) maintains the steam pressure below the lowest setting of the MSSVs during emergency shutdowns or hot standby conditions. Each of the two valves for a single steam generator is powered from a different Class 1E power source to meet the single-failure criterion for a main steamline break. The ADVs receive backup ac power from the gas turbine generator and can be are powered from the Class 1E batteries, with each valve powered from a separate channel. The ADVs can be operated manually from the control room or remote shutdown panel following a safe shutdown earthquake coincident with the loss of offsite power. The ADVs can be manually closed and positioned. These valves have both analog position and open/close indication lights. This meets the position in Branch Technical Position (BTP) RSB 5-1, "Design Requirements of Residual Heat Removal System," and in Issue 1 of NUREG-0138, "Staff Discussion of 15 Technical Issues Listed In Attachment To November 3, 1976 Memorandum From Director, NRR to NRR Staff." Additionally, ABB-CE, in response to RAI Q410.2 (June 30, 1988), stated that a safety-grade air supply will be provided to operate the ADV actuators, if the normal air supply becomes unavailable. The back-up air supply might be a nitrogen or Type A American National Standards Institute/American Nuclear Society-59.3 (ANSI/ANS) (1984), "Safety Criteria For Control Air System," air supply. This information was not reflected in CESSAR-DC Section 10.3.2.3. This was identified as a Confirmatory Item 10.3-1 in the DSER. In an amendment to the CESSAR-DC, ABB-CE stated in CESSAR-DC Section 10.3.2.3.2.3 that (1) the ADVs are manually operable from the main control room or the remote shutdown panel, and (2) diverse sources of motive and control power (125 Vdc supply from the station batteries) to the ADVs are provided to meet single failure criteria. The ADVs are designed with a return spring which causes the valve to fail closed on loss of motive power or loss of control signal. On this basis, Confirmatory Item 10.3-1 is resolved.

Following a steamline break, either all steam paths downstream of the MSIVs are isolated by their respective control systems following a main steam isolation signal (MSIS), or the results of a blowdown through a non-isolated path are shown to be acceptable. An acceptable maximum steam flow from a non-isolated steam path is 10 percent of maximum steam rate (8.6×10^6 kg/hr [19×10^6 lb/hr] at 6895 kPa [1000 psi] saturated steam conditions). As stated above, non-seismic portions of the system and other systems that may interact with the essential portions of the MSSS are designed to seismic Category II. Therefore, failure of non-seismic systems, structures, or components will not preclude operation of the safety-related portions of the MSSS.

Leakage detection for steam leakage from the system in the event of a steamline break is provided by the MSIS which is a part of the engineered safety feature actuation system (ESFAS). This system actuates on steam generator pressure and level and containment pressure (high) signals. The MSIS actuates the MSIVs, the MSIV bypass valves, and the isolation valves between the MSIVs and the steam generators to limit steam blowdown resulting from a steamline rupture or component malfunction. The system instrumentation and controls are evaluated in Chapter 7 of this report. ABB-CE revised CESSAR-DC ~~Section 5.4.5.3~~ to include up to a 5-second closure-time capability by the MSIVs to isolate the steam generators upon receiving a signal from the ESFAS. in 10.3

The system design conforms with the Commission regulations as given in GDC 2, 4, 5, and 34 as discussed above. Therefore, the staff concludes that the design of the MSSS conforms with the acceptance criteria of SRP Section 10.3 and is, therefore, acceptable.

and Feedwater System

10.3.6 Steam ~~Generator~~ Materials

secondary system

The staff evaluated the ~~steam generator~~ materials in accordance with SRP 10.3.6. The areas of review included selection and fabrication of materials, fracture toughness of Class 2 and 3 components, and ABB-CE's approach to erosion/corrosion. The open items in the draft SER were addressed as follows:

ABB-CE will comply with RG 1.71, "Welder Qualification for Areas of Limited Accessibility," except for the welder performance qualifications. ABB-CE's proposed alternative is to assign only the most skilled welders to weld those joints with limited accessibility. These joints are also subject to the inspections required by the applicable code. These precautions should provide adequate assurance of the acceptability of joints welded under conditions of limited accessibility. ABB-CE has provided an acceptable alternative to the recommendations in RG 1.71. On this basis, [redacted] is resolved.

are carbon steel or equivalent

In Amendment N, ABB-CE revised the CESSAR-DC to specify that stainless steel will not be used in the main steam, hot reheat, condensate, main feedwater piping systems, and the heater drain piping systems upstream of the drain control valves. However, extraction steam piping, heater drain piping downstream of the drain control valves, and other piping exposed to wet steam or flashing liquid flow will be chromium-molybdenum alloy steel, stainless steel, or equivalent. Stainless steel will be used as tubing in various heat exchangers and as cladding of tubesheets in low chloride environments with alloy content increasing with increasing chloride levels. For chloride levels above 800 ppm, high concentrations of dissolved solids (above 1000 ppm), or water contaminated by sewage discharges, titanium tubing will be used. These material selections have been used in numerous plants for twenty or more years and there have been no failures under the stated environments. Accordingly, ABB-CE has provided adequate assurance of the safety and structural integrity of these systems for the 60 year life of the plant. On this basis, [redacted]

In Amendment N, ABB-CE revised the CESSAR-DC and identified specific materials for the steam and feedwater system.

Carbon-manganese and chromium-molybdenum steels are to be used for the main steam and main feedwater systems. These specific grades are as follows:

Main Steam

ASME Class 2 Portion	SA 672 Gr. C70, Cl. 22	(>24" NPS)
	SA 106 Gr. B or C	(>24" NPS)
B31.1 Portion	A 672 Gr. C70, Cl. 22	(>24" NPS)
	A 106 Gr. B or C	(≤24" NPS)

Main Feedwater

ASME Class 2 Portion	SA 106 Gr. B or C	(≤24" NPS)
B31.1 Portion	A 672 Gr. C70, Cl. 22 or A 106 Gr. C	(>24" NPS)
add footnote from Pg 10.3-25	A 106 Gr. B or C	(≤24" NPS)

In Amendment U to the CESSAR-DC, ABB-CE addressed the issue of erosion degradation of piping systems. For carbon steel piping systems, the following methods to minimize erosion/corrosion are described in CESSAR-DC, Section 3.6.3.1.2.1:

1. The bulk velocity is limited to prevent excessive erosion of the pipe wall. The following velocity guidelines are used for carbon steel piping:

Recommended Bulk Velocity Guidelines

Service	Velocity
Steam Piping	45.7 m/sec (150 ft/sec)
Water	4.6 m/sec (15 ft/sec)
Recirculation Lines (Infrequent Use)	6.1 - 7.6 m/sec (20 - 25 ft/sec)

2. Piping design and routing will be utilized to lower susceptibility of the piping to pipe wall thinning.

ABB-CE proposed that engineering evaluations would be performed on a case-by-case basis utilizing industry accepted methods, such as the Electric Power Research Institute's (EPRIs) CHECKMATE. It was also noted that the feedwater and condensate system will be designed to avoid erosion damage.

ABB-CE's design and layout of piping systems considers the effect on the piping material from fluid velocity, bend location, and the location of flash points. The staff believes ABB-CE has demonstrated an appreciation of water characteristics, piping configuration, and materials selection, and that these factors will mitigate erosion/corrosion of piping systems. On this basis, [redacted] is resolved.

ABB-CE also provided, in Amendment N, a corrosion allowance value [illegible] 60-year plant design life and its technical basis for the determi[illegible] of that value.

Carbon steel is to be used for the main steam and main feedwater systems with a minimum corrosion allowance of 2.5 mm (.1 in.). This corrosion allowance is greater than that used in the design of most current plants and should account for the 60-year design life of a System 80+ plant. For piping sections which are more susceptible to erosion/corrosion degradation, chromium-molybdenum or stainless steel are used per CESSAR-DC, Section 10.3.6.2.G.5. On this basis, DSER Open Item 10.3-6 is resolved.

delete [illegible]

ABB-CE specified that corrosion/erosion-resistant materials be used in piping susceptible to corrosion/erosion.

The primary method of corrosion/erosion mitigation is prevention in the design stage. These design features include:

- Fluid velocity limits (CESSAR-DC, Sections 10.3.2.3.1.D and 10.4.7.2.5.K)

these were deleted per Section 3.6.3.1.2.1

Repeated from Pg

- Chrome-molybdenum or stainless steel are used in highly susceptible piping systems with flashing or two-phase flow (CESSAR-DC, Section 10.3.6.2.G.5)

The inspection programs described in the CESSAR-DC, Section 10.3.4 will provide further assurance that susceptible piping will not be excessively degraded by corrosion/erosion before it is detected. ABB-CE's inspection programs rely on the EPRI NP-3944, "Erosion/corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines," for guidance, which has been accepted by the staff. [redacted]

The design of System 80+ components will also address the potential influence of environmental effects on the fatigue life of materials over the 60 year design life. Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (Refs. 1 through 3). The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate.

The issue of environmental effects on fatigue is currently under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the pressure vessel research council (PVRC). These activities were initiated based on requests from the ASME B&PV code committee and the board on nuclear codes and standards (BNCS). The charter of the PVRC steering committee is to provide guidance and direction related to determining the effects of light-water reactor service environments on the cyclic life properties of applicable materials. The steering committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current fatigue (S/N) curves should be appropriate for PWR environments. Since the BNCS is not in complete agreement with the Steering Committee's position, this issue has not been resolved. ABB-CE will continue to monitor the industry activities related to fatigue curves and future fatigue analysis methodology.

System 80+ components will be designed to ASME B&PV Code rules. If the influence of environmental effects has not been incorporated into the Code rules at the time of the design, the potential effects will be addressed based on the technical understanding of the materials data and anticipated operating conditions. The commitment to account for the effects of the environment in the fatigue analyses, based on ASME Code requirements, has been added to the CESSAR-DC, Section 10.3.6.2 by Amendment N. On this basis, D[redacted]

During the development of the draft SER, the staff raised the issue that carbon steel materials may be susceptible to the mechanism of dynamic strain aging (DSA), which reduces the material fracture properties (Ref. 4). The NRC staff held several discussions with ABB-CE on this issue. ABB-CE, in its January 20, 1993 submittal, noted that industry studies are in progress to investigate the susceptibility of materials to DSA and the extent to which the fracture toughness properties are affected. The materials used in nuclear power plant engineered safety systems (carbon and alloy steels, stainless steels and nickel base alloys) have adequate fracture toughness, either as an inherent property of the material (austenitic stainless steels, and nickel alloys) or by ASME Code required Charpy v-notch testing of carbon and low-alloy steels. These materials have been extensively used in existing nuclear power plants and have performed successfully without failure. The staff concludes that these materials, by meeting the ASME Code requirements, will have an acceptable level of fracture toughness to account for DSA. On this basis, D[redacted] is resolved.

No copper alloys are used for components that are in contact with feedwater, steam, or condensate.

Oxygen-induced corrosion is minimized by using corrosion-resistant materials for the steam reheaters, feedwater heaters, and the condenser.

Main steam piping, hot reheat piping, condensate piping, feedwater piping, and heater drain piping upstream of the drain control valves are carbon steel.

repeat

repeat

Extraction steam piping, heater drain piping downstream of the drain control valves, and other piping exposed to wet steam or flashing liquid flow are chromium-molybdenum or stainless steel.

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR Part 50, §50.55a, GDC 1, "Quality Standards and Records," and 35, "Emergency Core Cooling," and Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. This conclusion is based on the following:

1. ABB-CE has selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME B&PV, and meet the requirements of Parts A, B, and C of Section II of the Code. ABB-CE, in [redacted] also met the recommendations [redacted] "Materials Code Case Acceptability ASME Section III, Division I," which describes acceptable code cases that may be used in conjunction with this industry standard.

2. The ASME Code imposes fracture toughness requirements for ferritic steel materials in Class 2 and 3 systems. The fracture toughness tests, chemical composition, and mechanical properties required by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating failure.

3. ABB-CE has met the requirements of RG 1.71 by meeting the regulatory positions or providing and meeting an alternative to the regulatory positions in RG 1.71 that the staff has reviewed and found acceptable. The onsite cleaning and cleanliness controls during fabrication satisfy the position given in RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

evacuation system (MCES). Heat is removed from the MC by the condenser circulating water system (CCWS). The MC system includes all components and equipment from the turbine exhaust to the connections and interfaces with the main condensate and other systems.

In the DSER, the staff stated that ABB-CE had not submitted system drawing(s) and a component design parameters table in CESSAR-DC Section 10.4.1. This was identified [redacted]n the DSER. In an amendment to the CESSAR-DC, ABB-CE provided a system description, representative design parameters and a reference flow diagram in CESSAR-DC Section 10.4.1, Table 10.4.1-1 and Figure 10.4.1-1 respectively. The classification of systems, structures,and components is provided in CESSAR-DC Table 3.2-1 by identifying the MC as non-nuclear safety (NNS) and non-seismic category, and indicating that the quality assurance requirements of 10 CFR Part 50, Appendix B, are not applicable. On this basis, ~~DSER Open Item 10.4.1-1 is~~ resolved.

In the event of a load rejection, the MC condenses up to 55 percent of the full-load main steam flow bypassed directly to the condenser by the TBS without tripping the reactor (refer to Section 10.4.4 of this report). If the MC is unavailable during a normal plant shutdown, a sudden load rejection, or a turbine trip, the spring-loaded safety valves can discharge full main steam flow to the atmosphere to protect the MSSS from overpressure. The MC removes non-condensible gases from the condensing steam through the MCES (refer to Section 10.4.2 of this report). The design also deaerates any drains that enter the condenser. Condenser tube material is Type 304L stainless steel tubing or equivalent for chloride levels below 200 ppm and 316L stainless steel tubing or equivalent for chloride levels up to 500 ppm, 904L stainless steel or AL-6X, or equivalent for chloride levels between 500-800 ppm, and titanium tubing or equivalent for greater than 1000 ppm of dissolved solids or chloride levels above 800 ppm.

The main condenser is initially tested in accordance with the heat exchanger institute standards for steam surface condensers. The condenser shells, hot wells, and waterboxes have access openings, and periodic visual inspections and preventive maintenance are conducted following normal industrial practice.

Discussion? below

During normal operation, startup and shutdown.

Although the design has radioactivity monitors in the system to detect leakage into and out of the MC during normal operation and shutdown, the main condenser has no radioactive contaminants inventory. Radioactive contaminants can only be obtained through primary to secondary system leakage due to steam generator tube leaks. Also, the steam generator blowdown system (SGBS) continuously samples the radioactivity of the steam generator blowdown (refer to Section 10.4.8 of this report and CESSAR-DC Table 9.3.2-1, which gives process sampling requirements for normal operation). Since the radioactivity is continuously monitored to detect leakage into and out of the condenser at the vacuum pumps discharge, 10 CFR Part 50 (Appendix A, GDC 60, "Control of Releases of Radioactive Materials to the Environment") is met with respect to failures in the design of the system which could result in excessive releases of radioactivity to the environment. The vacuum pump discharge is continuously monitored for radiation in order to detect steam generator primary-to-secondary tube leaks (refer to Section 10.4.2 of this report). The radiological monitoring capabilities are discussed in the Section 11.5 of this report. Also, the main condenser is non-safety-related, serves no safety function, and is not required for any safety shutdown. Flooding due to a failure in the MC (loss of water box or circulating water piping) is limited to the turbine building. High-level alarms in the turbine building sump alert operators in the event of leaks large enough to flood the building. The operator can isolate leakage paths and limit flooding. Additionally, no safety-related structure, systems, or components are located in the turbine building. Early leak detection is also provided in the MC system (refer to Section 3.4.1 of this report for a discussion of flooding).

? This is not for flooding (I believe you're talking about tubes leaks, 3 or 4 10.4.1.2H.1

The MC design conforms with the Commission regulations as given in GDC 60 as related to the failures in the design of the system which do not result in excessive releases of radioactivity to the environment and do not cause unacceptable condensate quality, or flooding of areas housing safety-related equipment, as discussed above. Therefore, the staff concludes that the design of the main condenser conforms with the acceptance criteria of SRP Section 10.4.1 and is, therefore, acceptable.

10.4.2 Main Condenser Evacuation System

The MCES is a non-safety system, located in the turbine building. The MCES removes air and other noncondensible gases from the condenser and maintains adequate condenser vacuum for proper turbine operation during startup and normal operation. The MCES includes equipment and instrumentation to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The major components of the MCES are the vacuum pumps that are used to create a vacuum on the MC. The system equipment is NNS, non-seismic category, and not subject to the quality assurance requirements of 10 CFR Part 50, Appendix B.

In the DSER, the staff stated that ABB-CE had not provided sufficient details (such as a detailed system description, a flow diagram, a piping and instrument diagram, and a tabulation of the design parameters of the system components) for the staff to complete its review. ABB-CE gave little system information for the MCES; missing information included such items as the number of vacuum pumps and the manner in which the system is operated. The lack of the above system information was identified [redacted] Item 10.4.2-1.

In an amendment to the CESSAR-DC, ABB-CE provided a system description and a reference flow diagram in CESSAR-DC Section 10.4.2 and Figure 10.4.2-1 respectively, and classification of systems, structures, and components in CESSAR-DC Table 3.2-I for the MC. The four packaged/skid mounted vacuum pumps (one spare) are used for hogging and holding modes of operations. The hogging mode reduces the MC pressure to 13 to 25 mm (5 to 10 in.) of Hg. The absolute and the holding mode reduces MC pressure to its operating value. Each vacuum pump is sized for 34 m^3/min [1200 standard cubic feet per minute (scfm)] for hogging and 1.4 m^3/min (50 scfm) for holding modes in accordance with heat exchanger institute standards for the surface condensers. On this basis, DSER [redacted] is resolved.

system
The design was originally based on the use of air ejectors.

In amendments to CESSAR-DC Section 10.4.2, ABB-CE stated that there was no direct connection between the main vacuum system and the reactor coolant system. Therefore, the normal function of one did not directly affect the other. The steam jet air ejectors (SJAEs) discharge was continuously monitored for radiation to detect steam generator primary-to-secondary tube leaks. However, the provisions for monitoring the discharge of the vacuum pumps might not discussed since these pumps might be used at different times than the SJAEs and may have a different discharge path. From the information in CESSAR-DC Section 10.4.2, the vacuum pump discharge path and/or the monitoring capabilities associated with this path were not identified. This was identified as DSER Open Item 10.4.2-2 in the DSER. In an amendment to the CESSAR-DC, ABB-CE provided a system description and a reference flow diagram in CESSAR-DC Section 10.4.2 and Figure 10.4.2-1 respectively, that shows that vacuum pumps, not SJAEs, are used for system hogging and holding operations. The vacuum pump discharge is continuously monitored and routed to a unit vent in the nuclear annex. The radiological monitoring capabilities are discussed in the Section 11.5 of this report. On this basis, [redacted] resolved.

The requirement of Commission regulation (1) GDC 60 as it relates to the MCES design for the control of releases of radioactive materials to the environment, and (2) GDC 64, "Monitoring Radioactivity Releases," as it relates to the MCES design for the monitoring of releases of radioactive materials to the environment, are considered met if the regulatory positions contained in the following RGs and industrial standard are conformed to:

(1) RG 1.26, "Initial Test Programs for Water-Cooled Nuclear Power Plants," as it relates to the quality group classification for the MCES that may contain radioactive materials, but are not part of the reactor coolant pressure boundary and are not important to safety.

(2) RGs 1.33, "Quality Assurance Program Requirements (Operation)," and 1.123, "Quality Assurance Requirements for Control of Procedurement of Items and Services for Nuclear Power Plants," as they relate to the quality assurance programs for the MCES components that may contain radioactive materials.

(3) "Standards for Steam Surface Condensers," as it relates to the MCES components that may contain radioactive materials.

In the DSER, the staff stated that ABB-CE had not demonstrated such conformance to the regulatory positions for the staff to conclude that the system is acceptable. This was identified as [redacted] in the DSER. In an amendment to the CESSAR-DC, ABB-CE stated in CESSAR-DC Section 10.4.2.1 that the system conforms with the guidance of RGs 1.26 and 1.28, and the hydraulic institute standards for steam surface condensers to satisfy the requirements of GDC 60 and 64. As identified in CESSAR-DC Table 1.8-1, RG 1.33 is not applicable and RG 1.28 is applied instead of RG 1.123 to MCES. On this basis, [redacted].

The MCES is in compliance with the requirements of GDC 60 and 64 with respect to the control and monitoring of releases of radioactive materials to the environment by providing a controlled and monitored MCES. The system meets the acceptance criteria of SRP Section 10.4.2 and is, therefore, acceptable.

10.4.3 Turbine Gland Sealing System

main and intercept or main and intercept

The turbine gland sealing system (TGSS) includes the equipment and instruments that are a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The system prevents steam from leaking out and air from leaking in through the turbine shaft glands and through various steam valve stems. The TGSS provides a continuous supply of "clean" steam from the main and auxiliary steam systems to the main turbine shaft seal, control valves, stop valves, intercept valves, and bypass valves. turbine This sealing steam is provided by the auxiliary steam system during cold startup or emergencies. Once the steam generators are brought up to full pressure, the sealing steam source is switched from the auxiliary steam source to the main steam source and or then to the high-pressure packings leakage source as the turbine is brought up to load during normal operation. Excess steam is discharged to the MC. The sealing steam keeps air from leaking into the steam cycle and potentially radioactive steam from leaking out of the steam cycle into the turbine building. The system returns the air-steam mixture to the turbine gland steam packing exhauster/condenser. The gland steam condenser, acceptable. This was identified as D[redacted]DSER. In an amendment to the CESSAR-DC, ABB-CE stated in CESSAR-DC Section 10.4.3.1 that the system conforms with the guidance of RGs 1.26 and 1.28 to satisfy the requirements of GDC 60 and 64. As identified in CESSAR-DC Table 1.8-1, RG 1.33 is not applicable and RG 1.28 is applied instead of RG 1.123 to TGSS. On this basis, DSER [redacted]d.

The turbine gland seal system meets the requirements of GDC 60 and 64 with respect to the control and monitoring of releases of radioactive materials to the environment by providing a controlled and monitored TGSS. The system meets the acceptance criteria of SRP Section 10.4.3 and is, therefore, acceptable.

10.4.4 Turbine Bypass System

The TBS bypasses up to 55 percent of the total full-power main steam flow at normal full-power steam generator pressure 6895 kPa (1000 psia). The TBS is a non-safety system and is located in the turbine building.

full turbine

control NSSS thermal ... to

In conjunction with the reactor power cutback system, the bypass capacity is intended to allow for a load rejection of any magnitude without tripping the reactor or lifting primary or secondary safety valves. The TBS also controls NSSS thermal conditions when reactor power exceeds turbine power or prevents the opening of safety valves following a unit trip. The TBS can maintain the NSSS in zero power conditions. In response to [redacted] (dated June 30, 1988), ABB-CE stated that the system bypass capacity is consistent with reactor transient analysis (55-percent capacity input). The system limits pressure limiting control during the loss of one out of three feedwater pumps and transmits a control element assembly automatic motion inhibit signal when the turbine power and reactor power fall below selected thresholds. It also provides for manual control of reactor coolant system temperature during a NSSS heatup or cooldown. The system is intended to enable operation of the turbine bypass valves in a manner that minimizes valve wear, maintains valve controls, and limits the flow imbalance between condenser sections to the flow capacity of one valve when all turbine bypass valves and condenser shells are available. It also in provides a redundant means of avoiding the excessive

The system provides for operation of the turbine bypass valves to

... 10.4-1 item 2

release of steam as the result of operator error or a single-component failure. The TBS provides a condenser interlock that blocks turbine bypass flow when unit condenser pressure exceeds the preset limit. The system is not required for the safe shutdown of the reactor and does not perform a safety function. The system description, system design and performance characteristics, and flow diagrams are provided in CESSAR-DC Section 10.4.4, Table 10.1-1, and Figures 10.1-2 and 10.3.2-1 respectively. The classification of systems, structures, and components are provided in CESSAR-DC Table 3.2-1 for the TBS.

OK

The TBS takes steam from the main steam header upstream of the turbine main stop valves and discharges it directly to the MC, bypassing the turbine generator. The system comprises all components and piping from the branch connection at the main steam system to the MC. The system consists of eight air-operated turbine bypass valves located in groups of four on the two steamlines, and associated piping and instrumentation. Normally, these valves are controlled by the steam bypass control system (SBCS) (refer to Section 7.7.1 of this report) but can also be controlled remotely or manually (local manual operation). The maximum capacity for a turbine bypass valve is approximately 10 percent of full steam flow at 6895 kPa (1000 psia). The turbine bypass valves fail closed to prevent uncontrolled bypass of steam to the condenser. The system valves modulate fully-open or fully-closed in a minimum of 15 seconds and a maximum of 20 seconds while in the automatic operation mode and are designed to open in less than 1 second and close in 5 seconds in response to SBCS signals. The system can control steam flow as low as approximately 3.3 percent of total full-power steam flow during hot standby to permit pre-core hot function testing. The bypass control system is addressed in Section 7.7.1 of this SER.

Should the bypass valves fail to open, the secondary side ASME Code safety valves provide ultimate steam generator overpressure protection and main steamline overpressure protection. Should the condenser (ultimate heat sink) not be available ~~during normal operation~~, an interlock will prevent opening, or if opened, will close the system valves and, thus, operation of the system has no adverse effects on the environment. The total system valve capacity (55 percent of total full-power steam flow at 6895 kPa (1000 psia), in

Reactor Coolant System

On this basis, the staff concludes that the system design meets the requirements of GDC 4 and 34, and that the TBS conforms with the applicable criteria of SRP Section 10.4.4 and can perform its intended function. The system design is, therefore, acceptable.

to here

10.4.5 Condenser Circulating Water System

Although the majority of the CCWS is a non-safety and an out-of-scope system provided by the COL applicant referencing the System 80+ certified design, the parts of the system that are located in the turbine building are within the certified design scope. The referenced CCWS provides cooling water for the turbine condensers and rejects heat to the environment via the normal heat sink. (A cooling tower is one of the means of rejecting heat for the reference plant. Site-specific considerations may allow for alternative means heat rejection ~~of transferring heat to the ultimate heat sink.~~) The CCWS comprises all components and equipment necessary for supplying the MC with a continuous supply of cooling water. The system description, representative design parameters and reference flow diagram are provided in CESSAR-DC Section 10.4.5, Table 10.4.5-1 and Figure 10.4.5-1 respectively. The actual design and operating parameters will be evaluated on a site specific basis by the COL applicant referencing the System 80+ certified design. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 by identifying the CCWS as NNS and non-seismic category, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are not applicable. The system is located partly in the yard and partly in the turbine building, and these locations are non-seismic category areas. The CCWS is not required to perform any safety function.

ABB-CE has provided interface requirements (IRs) in CESSAR-DC Section 10.4.5.1.2 to ensure adequacy with the System 80+ certified design. The system design minimizes the potential for water hammer by providing adequate filling and high point venting, and valve opening/closing times are selected to minimize water hammer effects. The reference design of CCWS is tubes and impurities that could enter the system through a condenser circulating water tube leak. The CCS is not required to perform any safety-related function, but is important in maintaining the secondary coolant quality.

lead cation with downstream

The system utilizes side-stream, full-condensate-flow polishers consisting of deep bed, mixed resin ion exchangers. The system is sized to maintain water chemistry within specified limits during continuous plant operation, assuming a condenser leak of 0.00006 L/s (0.001 gpm) and 0.006 L/s (0.1 gpm) during an orderly plant shutdown not to exceed 8 hours. Ion exchange resin regeneration or replacement provision is provided in the system. The demineralizer outlet lines have individual flow-regulating valves. The design permits full-flow recirculation and return to condenser hotwell deaerating sections for cleanup and verification of resin bed performance after resin replacement.

After reviewing the ABB-CE's proposed design criteria and design bases for the CCS and the requirements for operation of the system, the staff concludes that the design of the CCS and supporting systems is acceptable and meets the primary boundary integrity requirements of GDC 14, "Reactor Coolant Pressure Boundary." This conclusion is based on ABB-CE having met the requirements of GDC 14 as it relates to maintaining acceptable chemistry control for secondary coolant during normal operation and anticipated operational occurrences by reducing corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of reactor piping failures and of primary-to-secondary coolant leakage. ABB-CE's design of the CCS meets the requirements of BTP MTEB 5-3, "Monitoring of Secondary Side Water in PWR Steam Generators."

10.4.7 Condensate And Feedwater Systems

The CFS return condensate from the condenser hotwells to the steam generators. The condensate system is located in the turbine building, and the feedwater system is located in the turbine building, above the yard, in the nuclear annex, in the MSVHs and in the containment. The systems include all components and equipment from the condenser outlet to the connection with the NSSS and to the heater drain system, secondary water makeup system, and auxiliary feedwater system connections. The major components of the systems include the three condensate pumps, a deaerator storage tank, condensate

emergency

polishers, a gland seal condenser, low-pressure feedwater heaters, three feedwater booster pumps, three feedwater pumps, high-pressure feedwater heaters, feedwater regulating valves, the MFIVs and associated piping, valves, instrumentation and controls. All pumps are motor driven and are 50-percent capacity with the intent that two of the three condensate pumps and all three feedwater booster and feedwater pumps will be operating during normal power operation.

The systems also have a motor-driven startup feedwater pump that is utilized for startup and shutdown and is capable of delivering up to 5 percent of full feedwater flow to both steam generators in addition to pump recirculation requirements. The startup feed pump takes suction from either the condensate storage tank or the deaerator storage tank and injects water into the main feedwater lines upstream of the feedwater regulating valves. The NPSH available for the startup feedwater pump is greater than the required NPSH in either suction source alignment.

On a loss of feedwater and reactor trip, the startup feedwater pump can be started manually to maintain steam generator level. On a loss of offsite power with a turbine trip, the startup feedwater pump will be supplied power from the alternate ac source (gas turbine generator). For each steam generator, one feedwater regulating valve is provided to control feedwater flow to the economizer nozzles and one regulating valve on each steam generator is provided to control feedwater flow to the downcomer nozzles upstream of each feedwater regulating valve. A motor operated isolation valve is provided for isolation of each line to the steam generators. The system descriptions, design and performance characteristics, and flow diagrams are given in CESSAR-DC Section 10.4.7, Table 10.1-1, and Figures 10.1-1, 10.4.7-2, 10.1-2, and 10.4.7-1, respectively. The classification of systems, structures,and components is provided in CESSAR-DC Table 3.2-1

Except for that portion of the feedwater piping used by the EFWS, the only safety-related function performed by any portions of the CFS is containment isolation. The non-safety portions of the systems are designed to Safety Class NNS, non-seismic requirements. As identified in CESSAR-DC Table 3.2-1, the valves, piping, and associated supports and restraints of the main

[redacted]

feedwater (downcomer) line full of water. The System 80+ feedwater system and steam generators utilize both a downcomer and economizer feedwater line. Each steam generator economizer and downcomer feedwater line nozzle is equipped with a 90-degree elbow, and check valves upstream of the feedwater line connections to the steam generators (refer to CESSAR-DC Figure 10.4.9-1 and Section 10.4.9.1.2). The requirement for tests to verify that unacceptable feedwater hammer will not occur using plant procedures is incorporated into CESSAR-DC Chapter 14. In a letter of January 24, 1992, CESSAR-DC responded to staff ~~Q410.124~~, committing to more fully incorporate statements addressing water hammer in CESSAR-DC Sections 10.4.7.2.5, 10.4.7.2.6, and 10.4.7.2.7, and especially to address the requirements for incorporating water hammer concerns into plant operating and maintenance procedures, system pipe routing, and selecting stroke times for the feedwater regulating valves. Incorporation of this information was identified as OK ~~Confirmatory Item 10.4.7-1 in~~ the DSER. In an amendment to the CESSAR-DC, ABB-CE stated that (1) the COL applicant will make provisions for avoidance of water hammer including the development of system operating and maintenance procedures, (2) a 90-degree elbow facing downward is attached to each steam generator feedwater nozzle which would aid in the prevention of water hammer and deaerator and connected piping is designed to prevent water hammer and (3) main feedwater pipe routing design and the feedwater regulating valve stroke times are such that water hammer is precluded. ABB-CE has adequately addressed feedwater control valve and controller designs with respect to water hammer potential and has added a COL item ~~committed~~ to review operating and maintenance procedures to ensure that precautions taken will minimize or eliminate water hammers. On this basis, [redacted] Item 10.4.7-1 is resolved.

the dark markouts are OK.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5 with respect to the capability of shared structures, systems, and components to perform their required safety functions. ~~This was identified as COL Action Item 10.4.7-1 in the DSER.~~ Upon further review, the staff has determined that

In the DSER, the staff stated that ABB-CE had not included the functional testing of the systems and components to ensure structural integrity and leak-tightness, operability and performance of active components, and the capability of the integrated system to function as intended during normal, shutdown, and accident conditions. Therefore, compliance with GDC 46, "Testing of Cooling Water System," regarding functional testing of the CFS and its components, was identified as D~~SER Open Item 10.4.7.2~~. In an amendment to the CESSAR-DC, ABB-CE stated in Section 10.4.7.4 that the CFS testing includes the functional testing of the systems and components to ensure structural integrity and leak-tightness, operability and performance of active components, and the capability of the integrated system to function as intended during normal, shutdown, and accident conditions. On this basis, DSER Open Item ~~10.4.7.2~~ is resolved.

In a letter of January 24, 1992 (response to staff [redacted] ABB-CE stated that the completion of detailed plant operating and maintenance procedures is outside the scope of design certification; however, a statement was added to the CESSAR-DC requiring that adequate provisions for avoidance of water hammer be provided in developing plant operating and maintenance procedures. The staff agrees with this approach since the referenced procedure is site dependent. In an amendment to the CESSAR-DC, ABB-CE stated that the COL applicant shall develop the system operating and maintenance procedures for avoidance of water hammer for NRC review. T[redacted].

The design of the CFS and supporting systems conforms with the Commission regulations given in GDC 2, 4, 5, 44, 45, and 46. The design of the CFS meets the guidance of SRP Section 10.4.7, and is, therefore, acceptable.

10.4.8 Steam Generator Blowdown System

The SGBS controls the quality of water on the shell side of the steam generators by removing chemical impurities and radioactive materials which accumulate as a result of primary to secondary and condenser tube leaks and corrosion of the steam generator materials. A continuous high-flow blowdown controls the concentration of these impurities.

either .2% or

and/

Each steam generator has its own blowdown line with the capability of blowing down the hot leg or the economizer regions of the steam generator shell side or blowing down both. The system accommodates a continuous blowdown of up to 1 percent of the maximum steam flow rate and up to 10 percent for a short period of time, not exceeding 2 minutes. The blowdown fluid passes to a flash tank where the flashed steam is separated and directed to the low-pressure feedwater heaters. The liquid portion is processed by passing it first through a heat exchanger and then through a blowdown filter where the major portion of solid impurities is filtered out. After filtration, the blowdown fluid is directed to cation and mixed-bed demineralizers where ionic species are removed. The processed fluid, which should meet the secondary water chemistry specifications, is then returned to the condenser. The system continuously processes the blowdown fluid at a rate of 0.2 percent of each steam generator's maximum steam flow, for full-power operation and normal steam generator chemistry, and 1 percent when the chemistry is outside the normal limits. There is a provision to isolate the portion of the blowdown system exiting the containment by the redundant blowdown line isolation valves that would close upon a MSIS, containment isolation signal, or emergency feedwater actuation signal.

The staff reviewed the design for the SGBS in accordance with SRP Section 10.4.8. The scope of review included a piping and instrument diagram, seismic and quality group classifications, design process parameters, and ABB-CE's evaluation of the proposed system operation.

The SGBS design meets the primary boundary material integrity requirements of GDC 14 as it relates to maintaining acceptable water chemistry control during normal and anticipated operational occurrences by reducing corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of primary-to-secondary coolant leakage.

The SGBS is seismic Category I and ANS Safety Class 2 (which is equivalent to Quality Group B) from its connection to the steam generator inside the primary containment up to and including the first isolation valve outside the containment in accordance with RGs 1.26 and 1.29, since this portion of the SGBS is considered an extension of the primary containment. The SGBS downstream of the outer containment isolation valves is not safety-related and not seismic Category I, and meets the quality standards of Position C.1.1 of RG 1.143. The SGBS meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

portions of the

ABB-CE revised the CESSAR-DC to specify that the blowdown piping material will be made of stainless steel or other corrosion resistant material. On this basis, DSER Open Item 10.4.8-1 is resolved.

with potential exposure to two phase blowdown fluid

10.4.9 Emergency Feedwater System

The EFWS is an independent, dedicated, safety-related means of supplying secondary-side, quality feedwater to the steam generators for heat removal and preventing reactor core damage during emergencies. The system is designed to actuate automatically or manually in the event of a loss of normal feedwater including a loss of normal onsite and offsite power. The system can initiate, following a major LOCA, with operator action to keep the steam generator tubes covered to minimize potential containment bypass leakage with a pre-existing condition of primary to secondary leakage. The system is located within the Nuclear Island (NI) structures, including the containment, reactor building, and nuclear annex. During normal plant operation, the system has no function; the separate non-safety startup feedwater system with a non-safety-grade startup feed pump is used for normal startup and shutdown (refer to Section 10.4.7 of this report).

The EFWS includes all safety-related equipment from the emergency feedwater tanks to the connections with the steam generators. The EFWS contains two emergency feedwater storage tanks (EFWSTs). Each of the EFWSTs is intended to supply secondary coolant to one steam generator, although the capability to cross-connect the two EFWSTs by non-safety grade piping with a non-safety grade gravity feed condensate source for makeup is provided. Each EFWST is connected to one of the two steam generators through independent divisions of the EFWS consisting of two parallel pump lines. These pump lines discharge to a common header that injects into the main feedwater downcomer lines inside the containment as the last connection before the lines enter the steam

Comments on the System 80+ FSER Chapters 11 and 12

Chapter 11 FSER

Section	Description/Comment
11.1	Correct the 5th line to read "condensate polisher regeneration..." to be consistent with the terminology used in the CESSAR.
11.2.1, pg 11-6	Add the word "radwaste" before the words "control room" to clarify which control room the alarms are being provided in the 5th sentence.
11.2.1, pg 11-9	Use consistent terminology when referring to the condensate cleanup system polishers. Replace the word "demineralizers" with "polishers".
11.2.1, pg 11-8	Add words "cleanup system polisher" between condensate and regenerant for clarification.
11.3.1, pg 11-19	The first word of the 6th line should be "evacuation" not "evaluation".
11.4.1, pg. 11-27	Again, change "demineralizers" to "polishers" to be consistent with the CESSAR terminology.
11.4.1, pg. 11-27	Correct the 10th line to read "A spent resin decanting tank...".

Chapter 12 FSER

Section	Description/Comment
12.1, pg 12-4	The 8th line states that "electrical components containing radiation-sensitive materials will be shielded or located in low-radiation areas" does not appear in the CESSAR-DC; however, it does appear in the System 80+ ALARA Guidelines Manual. This statement will be added to Section 12.1 of CESSAR-DC to be consistent with the FSER.

Chapter 12 FSER(Cont'd)

Section	Description/Comment
12.1, pg 12-5	The 1st line states that "Valves located in high-radiation areas will be equipped with reach rods or motor operators to minimize radiation exposure." does not appear in the CESSAR-DC in that exact wording. Section 12.0 states "radiation protection measures include: ... use of remotely operated valves or handwheel extensions". The wording used in the FSER does appear in the System 80+ ALARA Guidelines Manual. This statement will be added to Section 12.3 of the CESSAR-DC.
12.1, pg 12-5	The 8th line states "The System 80+ design will minimize the use of evaporators ..." does not appear in the CESSAR-DC; however, it does appear in the System 80+ ALARA Guidelines. This statement will be added to Section 12.1 of the CESSAR-DC.
12.1, pg 12-6	The 5th and 6th lines state "Equipment such as sound-powered telephones or closed-circuit television will be used during high-dose jobs..." The phrase "high-dose jobs" is not consistent with Section 12.1.3.B of the CESSAR-DC. This section uses the phrase "long-duration jobs". This inconsistency should be corrected in the FSER.
12.3.1, pg 12-12	The 9th line states "Mechanical snubbers rather than hydraulic snubbers will be used in radiation areas ..." does not appear in the CESSAR-DC; however, it does appear in the System 80+ ALARA Guidelines. This statement will be added to the Section 12.3.1 of the CESSAR-DC.
12.3.1, pg 12-13	The 11th line of the 1st complete paragraph states "Crud traps in welds will be minimized by using butt welds in lieu of socket welds." does not appear in the CESSAR-DC; however it does appear in the System 80+ ALARA Guidelines. This statement will be added to Section 12.3.1 of the CESSAR-DC.

Comments on the System 80+ FSER Chapters 11 and 12

Chapter 12 FSER(Cont'd)

Section	Description/Comment
12.3.2, pg 12-19	The 12th and 13th line do not include the specification that there will be an electrical interlock between the area radiation monitor and the lockable access door to the incore chase to prevent access to the incore chase during withdrawal of the incore instrumentation. This statement should be added to ensure the FSER is consistent with the Section 12.3.2, Amendment U of the CESSAR-DC.

Statement must discuss source term flow. Surry does not have charcoal delay beds

as DSER Open Item 11.3-1 in the DSER. By submittal dated November 4, 1993, ABB-CE stated that the chosen value was based on Chapter 12 specifications for PWRs given in Electric Power Research Institute Utility Requirements Document for advanced light water reactors. ABB-CE further stated that operating experience at PWRs ~~that use charcoal delay beds for waste gas treatment~~ (e.g., Surry and Beaver Valley units) showed less than 0.0028 m³/m (0.1 scfm) average flow rate ~~through the delay beds~~. [illegible] at the higher chosen value was therefore conservative, since delay [illegible] varies inversely as the flow rate. The staff finds the above response satisfactory and so, DSER Open Item 11.3-1 is resolved. The staff also finds that the ABB-CE calculated delay times are in agreement with NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from PWRs," [illegible] 1 methodology. The process gas subsystem effluent is additionally processed, if required, by the nuclear annex filtration system, which consists of a pre-filter, high-efficiency particulate air (HEPA) filters and charcoal adsorber before it is discharged to the plant vent via the nuclear annex ventilation system. Thus by including charcoal delay beds, the process gas subsystem meets GDC 60 with regard to control of radioactive release to [illegible]tricted areas.

The GWMS is a non-safety-related system and has no accident mitigation functions. The process gas portion of the system is located in the nuclear annex, which is a seismic Category I structure and, therefore, designed to withstand a safe-shutdown earthquake (SSE). The process gas subsystem and the structure housing the subsystem are designed in accordance with the applicable Guidelines C.2, C.4, C.5, and C.6 of RG 1.143 with respect to specific guidelines for gaseous radwaste systems; general guidelines for design, construction, and testing criteria for radwaste systems; specific seismic design criteria for gaseous waste management system; general seismic design criteria for structures housing radwaste systems; and general guidelines for providing QA for radwaste management systems. CESSAR-DC Subsection 11.3.1.2 provides a detailed discussion of how the design of the subsystem and its housing structure meet the applicable guidelines of RG 1.143. Specifically, the subject section states that the GWMS (i.e., process gas portion) is designed radwaste building for eventual shipment to a licensed burial facility. The low-activity bag filters (which are on the discharge lines of low activity spent resin transfer pumps) that need replacement are lifted from their housing and placed in adjacent shielding containers (usually HICs; however, 55 gallon drums may also be used) by remote handling tools, after water is purged from the filter housing and the filter media is dewatered using compressed process air. The filled containers are moved to shielded storage area for eventual shipment to a licensed burial facility.

As stated in Section 11.2 of this report, regenerant demineralizers will be used in the condensate cleanup system. The secondary side resins from these demineralizers will be processed as necessary and packaged for disposal if the resins become physically broken or the DF is reduced. At this time, the resins will be sluiced into the shipping container and dewatered in the turbine building. HICs will be used only as necessary to ensure compliance with the Department of Transportation (DOT) regulations. The SG blowdown treatment demineralizer resins will also be processed in the turbine building similar to the condensate cleanup system resins; however, these resins will not be regenerated. A spent resin decaying tank of sufficient capacity is provided in the turbine building to facilitate processing of the secondary side resins, as necessary, based on sampling of the tank contents.

Dry solid wastes such as contaminated cloth, paper and plastic are compacted by a dry solid compactor in the low-level waste handling and packaging area. These and non-compatible dry wastes including HVAC system filter assemblies are packaged in 55 gallon drums or boxes and stored in a low-level solid waste storage area of the radwaste building for eventual shipment.

The spent resin tanks and the shipping containers are sampled and surveyed to verify that the dewatering or solidification (if required) and packaging are complete and meet the guidelines of BTP ETSB 11-3, which provides the design guidance for SWMS. Additionally, the wet solid wastes will be processed and disposed in accordance with 10 CFR Part 61 requirements. To ensure the above compliance, ABB-CE has identified a COL interface (CESSAR-DC, Section 11.4.1.1, Item F) which calls for the owner operator to process and classify the packaged waste in accordance with 10 CFR Part 61 requirements.

12.2.3 Sources Used in NUREG-0737 Post-Accident Shielding Review

The initial core releases that will be used to determine postaccident radiation levels will be equivalent to the source terms recommended in Draft NUREG 1465 RG 1.4, RG 1.7, and SRP Section 15.65 6.5. This is in accordance with 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660 and NUREG-0737). Item II.B.2 of NUREG-0737 states that applicants should identify systems that contain high levels of radioactivity in postaccident situations. Since ABB-CE does not have the specifications for either the "as-built" systems or the "as-procured" hardware that are needed for a completed plant, CESSAR-DC Section 12.2.3 does not contain a listing of such postaccident sources. This lack of information was identified in the DSER as Open Item 12.2.3-1. To address this open item, ABB-CE has provided DAC in Table 3.2-1 of ABB-CE's CDM. These DAC specify the methods and assumptions for determining the post-accident source terms and state that analyses will be performed to determine the radiation levels in areas which require access for mitigation of or recovery from a design basis accident. The methods and assumptions used to calculate the post-accident source terms in Section 12.2 and the methods and assumptions specified in the DAC are consistent with the SRP acceptance criteria and will ensure that the System 80+ plant meets the requirements of 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660 and NUREG-0737). DSER Open Item 12.2.3-1 is, therefore, resolved by ABB-CE's issuance of DAC 3.2. Evaluation of the DAC process is given in Section 14.3 of this report.

The DSER for Section 12.2 identified the following three Open Items; 12.2.1-1 (also discussed in Section 12.1), 12.2.2-1 (also discussed in Section 12.3), and 12.2.3-1 (also discussed in Section 12.3). All three of these open items were subsequently addressed and resolved by the issuance of DAC 3.2. On the basis of the above, the staff's review of Section 12.2 of CESSAR-DC is complete and there are no remaining outstanding open items.

12.3 Radiation Protection Design

The staff reviewed the facility design features, shielding, ventilation, and area and airborne radiation monitoring instrumentation described in CESSAR-DC

In the DSER, ABB-CE was requested 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 provided 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 reviewed the information and found that the application of the convolution method to the 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 ~~type~~ (including the spacer grid and components) that adversely ~~that may change the thermal-hydraulic characteristics) from that presently proposed, or uses correlations other than~~ impact the applicability of the CE-1 correlation for DNBR calculations, ABB-CE is required to provide 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's 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 Nuclear Regulatory Commission (NRC) in a letter dated August 28, 1990. The revised list should include all design deviations and should justify the adequacy of the deviations for System 80+. This was identified in ~~as~~ DSER Open Item 15.1-3. The staff has reviewed ABB-CE's responses addressing the System 80+ design deviated from the EPRI URD requirements and found that the responses are acceptable for closure of Open Item 15.1-3. The staff's evaluation for the closure of DSER Open Item 15.1-3 is included in Section 1.1 of this report.

Will Teipo be sensitive to this?

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

1. An SLB inside containment at full power with concurrent LOOP in combination with a single failure and a stuck CEA.

2. An SLB inside containment at full power in combination with a single failure and a stuck CEA.

3. An SLB inside containment at zero power with concurrent LOOP in combination with a single failure and a stuck CEA.

4. An SLB inside 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 with concurrent LOOP in combination with a single failure, a stuck CEA, and TS SG leakage. X

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 steam line 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 steam line, which is limited by the flow restrictor throat area, is approximately 30 percent of the steam line cross-section area, or 0.119 m^2 (1.28 ft^2).

<u>Initial Conditions and Analytical Assumptions</u>

Steam line 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 on low SG system pressure trip signal.

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 was previously approved by the staff 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 provided 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 will initiate 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 2.28×10^3 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 (Cases 2 and 5), and initiated by a low SG pressure trip signal for an SLB at zero power initial conditions (Case 4).

The analytical results demonstrated 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 identified as the limiting SLB for worst radiological consequences. The staff finds in the analytical results that Case 5 (an SLB outside containment during full-power operation with concurrent LOOP offsite power available 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 meet 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 report.

Since no fuel failure is predicted, the statistical convolution method was not used by ABB-CE 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.

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.

The FLB analysis assumed that the LOOP power 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 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. 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 indicated 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 seconds 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, (dated July 1987). 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 (2793 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 results show that the peak pressure of 1.85 x 10^4 kPa (2676 psia) is within 110 percent of the design pressure and demonstrate 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 the an SF and available offsite power is a low probability event.

ABB-CE's results of 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 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 of 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 (2,635 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 report, 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 report, the staff determines that this applicant's 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's review of the radiological releases is discussed in Section 15.4 of this report.

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 have been made to make this accident extremely unlikely, ABB-CE analyzes the consequences of such an event.

not exceed (1) the exposure guideline values set forth in 10 CFR Part 100 for both the preaccident iodine spike and fuel failure cases, and (2) a small fraction (i.e., 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 steam line failure outside containment.

As discussed in Section 15.1 of this report, 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.

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 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 which 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.

To be revised by CE

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 0.0~~65~~56 m^2 (0.~~7~~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 ~~57,700~~ 65,900 kg (~~127,000~~ 145,000 lbm) of steam was calculated to be released from the feedwater system to the atmosphere during the first thirty 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

In addition to the TS reactor coolant activity, the primary coolant 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-generated iodine spike is assumed.

the period between 30 minutes and 8 hours, ABB-CE assumed that steam releases are the same as for the steam line break case, since the cooldown is the same.

Two sources of activity were considered by ABB-CE 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. activity releases based on the initial activity in the secondary coolant as well as from activity associated with primary to secondary leakage 15

In ABB-CE's analysis, thyroid doses at the exclusion area boundary of 0.2 0.23 Sv (20 23 rem) were computed the worst case. ABB-CE also computed a whole body dose of 3.3 7.6 x 10^{-4} Sv (0.076) rem 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 X/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 set forth in SRP 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.

rotor event are within staff acceptance criteria as set forth in SRP 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 the SRP 15.3.3 as modified by applicable assumptions set forth 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 (viz., 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. ~~considered timing for releases arising from this accident, consistent with draft NUREG-1465 and as set forth in Section 15.A.7 of this report.~~ 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 Appendix A 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.

The greatest potential for offsite dose consequences for this event was determined by ABB-CE to be 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 where 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, viz., the activity available for leakage from the containment (in the first 30 minutes) 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." Specifically, ABB-CE considered the activity in the fuel pellet clad 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 clad gap is assumed to be instantaneously mixed throughout 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 has 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 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 CESSAR-DC design, are sufficient to provide 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/clad 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 ~~considered timing for releases arising from this accident consistent with draft NUREG-1465 and as set forth in Section 15.A.7 of this report~~. For additional information on Source Term Related Technical and Licensing Issues, refer to Appendix A 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 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/sec (27 lbs/sec) of primary coolant is released as a result of a double ended break of a letdown line outside containment, upstream of the letdown line control valve. In addition, ABB-CE noted that the maximum break flow, which is about one and one half times the expected letdown flow, is limited to 12.3 kg/sec (27 lbs/sec) by the use of letdown line orifices located inside 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

1. Steam generator tube rupture without a concurrent loss of offsite power,
2. Steam generator tube rupture with a concurrent loss of offsite power, and
3. Steam generator tube rupture with a loss of offsite power and a single failure.

Because no fuel failure is expected to occur as a result of a SGTR event under any of these conditions, ABB-CE assumed a three 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.

A 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 set point.

It was further assumed by ABB-CE that after 1800 seconds (i.e., thirty minutes) the operator initiates a plant cooldown using the unaffected steam generator, atmospheric dump valves, and the emergency feedwater system. In ABB-CE analysis, it was assumed that for releases ~~via the atmospheric dump valve~~ from ~~the~~ unaffected steam generator, a DF of 1 100 resulted for the iodines.

ABB-CE's analysis of the radiological consequences of a steam generator tube rupture event with a LOOP and a limiting single failure was reviewed by the staff. 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 a steam generator tube rupture accident do not exceed (1) 2 the exposure guideline (with the exception of event sequence ?

valves set forth in 10 CFR Part 100, and (2) 10 percent of these exposure guideline values for a 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 performing analyses of the radiological consequences of the spectrum of LOCAs in the CESSAR-DC, ABB-CE utilized the assumptions specified 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 are presented Appendix 15A to the CESSAR-DC.

Draft NUREG-1465 provides 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 Appendix A to this chapter as fractions of the total core inventory.

Releases were assumed by ABB-CE to be uniform over the duration given in Table 3.6 of draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants."

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. ABB-CE conservatively assumed that 0.25 percent of the iodine released was organic for purposes of the radiological analyses.

Containment sprays were assumed to operate to remove airborne radionuclides; this removed activity is 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, no credit was taken for either radioactive decay in transit or for ground deposition in transit.

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

1. Discharge of iodine spike activity contained in the reactor coolant.

2. 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.

3. 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 phases: (1) coolant release phase, (2) gap release phase, and (3) early in-vessel release phase.

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

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

Based on information contained in draft NUREG-1465, "Accident Source Term for Light Water Nuclear Power Plants," ABB-CE assumed the entire release was particulate (except for the noble gases and five percent of the iodines). Five percent of this five percent was assumed to be organic. ABB-CE also considered timing for releases arising from this accident consistent with draft NUREG-1465 and as set forth in Section 15.A.7 of Appendix A to this chapter. For additional information on Source Term Related Technical and Licensing Issues, refer to Appendix A to this chapter.

ABB-CE computed doses at the exclusion area boundary for releases during the first two hours and at the low population zone from releases over the assumed 30-day duration of the event. Total doses at a given location were computed by ABB-CE considering releases from the following release paths: (1) main steam safety valves, (2) ADVs, (3) nuclear annex, and (4) containment.

X (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 meet 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. The analysis included appropriate radionuclide sources and transport paths as described above.

The staff has 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.

The staff concludes, based on its review of the methods, assumptions, and parameter definitions used by ABB-CE, 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 load movement is performed as planned.

ABB-CE analyzed the radiological consequences of a FHA occurring in the containment and a FHA occurring in the fuel building. In performing its analysis, ABB-CE assumed operation of the containment purge ventilation system and associated filters for the FHA inside 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.

Offsite radiological consequences to the whole body from immersion and to the thyroid due to inhalation were computed by ABB-CE 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 containment and in the fuel building.

The staff concludes that the specified site parameters related to the exclusion area and low population zone, 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 meet 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

To be reviewed by DESI

Using an alternative process, ABB-CE concluded that the maximum allowable dilution factor is 3.55 x 10^{-6} $\boxed{1.49 \times 10^{-5}}$. This value reflects the minimum extent to which the radioactive liquid released from the failed BAST will be diluted prior to reaching the potable water supply. Based on 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 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 meet all applicable criteria, then the radiological consequences of a spent fuel cask drop accident need not be estimated.

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

Finally, ABB-CE noted that the cask handling crane is provided with 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 satisfy the applicable criteria of SRP 15.7.5, no radiological impact evaluation of a cask handling accident is required.

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

In Section 15 of the CESSAR-DC, ABB-CE presented the results of a dose calculation for a sequence which conservatively represents the systems and

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

15.A.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 evolutionary designs, such as the Systems 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 regarding fission product releases into the containment would be utilized.

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 which follow.

15.A.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, "Accident Source Terms for Light-Water Nuclear Power Plants," should be based on the use of gap release and the early in-vessel releases for design basis accident evaluations. The staff considers the inclusion of the late in-vessel and the ex-vessel source terms to be overly conservative for design basis accident (DBA) purposes. In essence, these events are of such low probability that they are not credible in terms

[1] Memorandum, James M. Taylor to The Commissioners, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs, dated February 10, 1994. [NUDOCS Accession No.______] ?

subsequent behavior after entering containment from the RCS. This report points out that, among other things, containment water exposed to air will absorb carbon dioxide to form carbonic acid. This would slightly lower the pH, 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.

Subsequent to the issuance of draft NUREG-1465 in June 1992, the staff issued NUREG/CR-5950, "Iodine Evolution and pH Control" in December 1992. This report points out that the most important acids formed in 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 chemical additive for pH control during the initial stage of a LOCA. ABB-CE stated that the pH of the water in the IRWST is maintained at a minimum of 7.0 to control post-accident evolution of elemental iodine and to minimize corrosion of the stainless steel in the containment. raised to a level

A total mass of 18,930 kg 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 elevation of the baskets is 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

accidents to include ex-vessel releases and late in-vessel releases. In considering the resultant radiation environment, the staff concluded 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 3 of this report.
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15.A.5 Iodine Deposition on Steam Lines and Condenser -- does not apply to System 80+

15.A.6 Fission Product Holdup in Secondary Containment -- does not apply to System 80+

15.A.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, this draft NUREG indicated that fission product early in-vessel releases were estimated to start no earlier than 0.5 hours for PWRs. As noted in 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 hours 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 NUREG-1465. ABB-CE's analysis was found to be conducted in accordance with staff guidelines, and is, therefore, acceptable.

15.A.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" (Ref. ~~4~~ 1). The EPRI report references the mechanistic correlation developed by the industry degraded core rulemaking program (IDCOR). The correlation establishes 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 , by doing so, corresponds to an airborne concentration of approximately 0.02 gm/m^3 in the correlation. It neither considered diffusiophoresis nor hygroscopicity which 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 1×10^5 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 (to be published as NUREG/CR-6189) under a contract with the Sandia National Laboratory. Major insights from that model were used by the staff to perform a comparative analysis with the model used by ABB-CE. The staff's model uses two 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. Based on these comparisons, the staff concludes that ABB-CE's model is adequately conservative, and therefore, the staff finds it to be acceptable.

15.A.9 Aerosol Removal by Suppression Pool -- **does not apply to System 80+**

15.A.10 Containment Spray Removal

GDC 41, 42, and 43 of Appendix A to 10 CFR Part 50 require systems to control fission products to 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 requirements document for evolutionary plant designs requires a CSS. ABB-CE has provided 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 liters (5000 gallons) per minute. 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 2.1×10^6 Liters (545,800 gallons). The CSS is designed to operate throughout the entire duration of a LOCA.

The total free volume of the System 80+ containment is 9.25×10^4 m^3 (3.34×10^6 ft^3) of which the effective spray volume is 7.67×10^4 (2.74×10^6 ft^3) (approximately 82 percent of the containment free volume). ABB-CE assumed the remaining 18 percent to be unsprayed. ABB-CE also assumed the average weighted fall height of spray droplets to be 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 (Ref. 1). 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 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 and 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 (Ref. 2) which is a further modification of the NAUA-4 code (Ref. 3) to include the effects of hygroscopicity on particle steam condensation and removal by diffusiophoresis.
ABB-CE, however, stated that the effects of hygroscopicity have not applied to the containment spray system performance evaluation for the System 80+ design.

In implementation of 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 simplistic model that can be used to estimate aerosol removal by sprays without the necessity of using detailed systems codes such 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. The staff finds the following:

1. The average spray droplet size of 1000 micrometers (μm) used by ABB-CE is more conservative compared to 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. The staff's model assumes that non-radioactive aerosols are produced (350 kg from in-vessel releases) while ABB-CE does not.

3

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

4

5. 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 that used by the staff.

6. ABB-CE used the diffusiophoretic capture of aerosols which is neglected .. the staff.

7. ABB-CE and 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. As a result, the staff finds ABB-CE's spray model proposed for the System 80+ containment design to be acceptable.

15.A.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 is contained in Section 6.4 of this report.

The annulus ventilation system consists of two redundant ventilation systems; each system consists of a fan, a filter train, associated ductwork, dampers, and necessary controls. The annulus ventilation system provided 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.

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

Model not used for System 80+ evaluation. Staff's analysis is contained in Section 6.4 of this report.

15.A.13 Failure of Passive Containment Cooling System -- does not apply to System 80+

15.A.14 References for Appendix A

1. "Licensing Design Basis Source Term Update for the Evolutionary Advanced Light Water Reactor," Advanced Reactor Severe Accident Program Source Term Expert Group, September 1990.

X 2. "SWNAUA VER.[02] LEV01[0], on a PC "Aerosol Behavior in a Condensing Atmosphere - Diffusiophoresis and Spray Version," NU-185, August 1986. May 1993

3. Bunz, H. et [illegible] NAUA Mod 4: A Code for Calculating Aerosol Behavior in LWR Cor[illegible]ents, Code Description and User's Manual, Preliminary Descri[illegible] 1982.

with the provisions of 10 CFR 52.45. In the initial application on March 30, 1989, ABB-CE applied for the SDC in accordance with Appendix O of 10 CFR Part 50. On August 21, 1989, ABB-CE revised its SDC application to be pursuant to 10 CFR Part 52. However, in the staff's letter dated May 1, 1991, the NRC stated that the contents of the ABB-CE application for System 80+ design were made in conformance with the requirements of 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, the applicable date for the appropriate supplement of NUREG-0933 for paragraph 52.47(a)(1)(iv) is six months prior to ABB-CE's submittal to be in conformance with 10 CFR 52.47 or the March 4, 1991, date. In Amendment U to Appendix A of CESSAR-DC, ABB-CE committed to address the relevant issues in Supplement 15 of NUREG-0933 dated ______. The applicable supplement of NUREG-0933 is therefore, Supplement 15.

The staff reviewed Supplement 15 of NUREG-0933 to determine 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 meet paragraph 52.47(a)(1)(iv). In addition, five other issues (A-17, A-29, B-5, 29, and 82) were added to the list. These were issues that were resolved without the issuance of new requirements, but for which the Office of Nuclear Reactor Research, NRC, had recommended the development of specific guidance for future plants.

The issues needed to meet paragraph 52.47(a)(1)(iv) are evaluated in Sections 20.1 to 20.4. Additional issues which ABB-CE considered applicable to the System 80+ design were included in Appendix A of CESSAR-DC 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 GSI that are technically relevant to the System 80+ design as required by 10 CFR 52.47(a)(1)(iv). Some of these items involve COL action items and will be the responsibility of the COL applicant.

Compliance with 10 CFR 52.47(a)(1)(ii)

Paragraph (1)(ii) of 10 CFR 52.47(a) requires an application for a standard design certification to include a demonstration of compliance with any technically relevant portions of the TMI action plan requirements in 10 CFR 50.34(f).

ABB-CE addressed the 50.34(f) TMI action plan requirements in Appendix A of CESSAR-DC. These requirements are discussed in Section 20.5 of this report. Due to the overlap between these TMI items and the TMI items from NUREG-0933 in Section 20.3 of this report, Section 20.5 lists all the 50.34(f) TMI items in tabular form. This provides the issue designation and a reference to the appropriate section in Section 20.3 of this report which contains the evaluation of the TMI 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 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 standard design certification process preserves operating experience insights in the certified design. As discussed in Section 20.6 of this report, the staff concludes that ABB-CE has adequately considered operating experience identified in generic letters and bulletins issued by the Commission since 1980 in the System 80+ design.
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Resolution of Issues Relevant to the System 80+ Design

In Section 1.0 of Appendix A of CESSAR-DC, ABB-CE listed the issues in Supplement 15 of NUREG-0933 that it considered relevant to the System 80+ design. The justification for ABB-CE considering an issue not relevant to the design was also provided in Section 1.0. The resolutions of the issues that

ABB-CE considered relevant to the design and that the staff considered relevant in terms of 52.47(a)(1)(ii) and (iv) are given in Sections 20.1 through 20.5.

There are also issues that were evaluated by the staff and are discussed in these sections which ABB-CE did not consider relevant in Amendment U and the staff does not consider relevant in terms of Supplement 15 of NUREG-0933. These issues were evaluated by the staff during the review of the System 80+ design since it was submitted by ABB-CE in 1989, and it was decided to keep the evaluations in Chapter 20 of this report.

Table 20.1 lists the relevant USIs and GSIs (i.e., issues) for the System 80+ design, the sections where the issues appear in this chapter, and the basis for the relevancy of the issue to the design. The relevancy of the issues are the following, as discussed above: (1) the issue is required by 10 CFR 52.47(a)(1)(ii) or (iv), (2) the issue was selected by ABB-CE as being relevant in CESSAR-DC Appendix A, and (3) the staff decided to discuss the issue. In the latter case, ABB-CE originally stated the issue was relevant in an early amendment to Appendix A and then concluded that the issue was not relevant to the System 80+ design. These latter issues and the evaluations by the staff were left in Chapter 20 of this report. The issues are arranged in the order they appear in Sections 20.1 through 20.4 of this chapter.

20.1 Task Action Plan Items

The task action plan items, except for A-48 and B-26, which are evaluated against the ABB-CE System 80+ design in this section were evaluated (1) for the design to meet 10 CFR 52.47(a)(1)(iv) and 10 CFR 50.34(f) and (2) because ABB-CE stated in Appendix A of CESSAR-DC that the task action plan item applied to the design. ~~ABB-CE has not adequately proposed resolutions for the following Task Action Plan Items: B-17, B-36, C-1, and C-17. ABB-CE has been requested to address these task action plan items.~~

Issue A-1: Water Hammer

Issue A-1, formerly USI A-1 in NUREG-0933, addresses the issue of water hammer generator tube rupture events are not dominant contributors to risk (Refs. 1 and 2). However, the staff issued a generic communication, GL 85-02, "Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity." 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 indicates that the staff's recommendations in GL 85-02 will be followed, and they are: prevention and detection of loose parts, steam generator tube inservice inspection, secondary water chemistry and impurity control, primary-to-secondary coolant leakage limit, primary coolant iodine activity limit, and safety injection signal reset logic. ABB-CE also specifies A-690 for SG tubes to provide increased resistance to corrosion.

The staff finds that ABB-CE's proposed resolution to Issue A-4 is acceptable. The staff has recently initiated rulemaking to address more recent experience with steam generator operation. ABB-CE will be subject to the applicable requirements of any rule that is promulgated in this area.

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 meet the recently published Electric Power Research Institute (EPRI) guidelines for makeup water to steam generators. Therefore, DSER Open Item 5.4.2-5 is resolved.

The CESSAR-DC specifies that development of the steam generator tube inservice inspection 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 inservice inspection program is identified as COL Action Item 20.1-1.

See Issues 66 and 135 for additional evaluations of steam generator issues.

A nuclear power plant comprises numerous systems, structures, and components (SSCs) that are designed, analyzed, and constructed by 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 USI 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 USI A-17 program, the staff requested, as stated in NUREG-0933, that plant designers consider the operating experience as 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 the staff RAI Q440.127(1) listed in Appendix B of this report, ABB-CE stated that the System 80+ plant is designed to prevent ASIs from water intrusion, internal floods, seismic events, and pipe ruptures. ABB-CE gave examples of the design features to prevent ASIs. In the resolution to USI A-17 included in CESSAR-DC Appendix A, ABB-CE indicated that the System 80+ design is 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," 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 of issuance of the DSER, ABB-CE was scheduled to revise the System 80+ PRA. As part of the scheduled revision there were 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 evaluating induced-human-intervention-coupled ASIs in parallel with the System 80+ PRA revision. In addition, ABB-CE committed to provide an inspections, tests, analyses, and acceptance criteria (ITAAC) program acceptable to the NRC for ASI risk 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. full-flow

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 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 provided at the entrance of the HVT to prevent large debris from entering the HVT and thus the IRWST. These vertical screens are greater than 6 feet high and more than forty feet long. The HVT is of sufficient volume as to allow significant settling of high density debris. ~~to occur.~~

The fine debris that ~~is~~ 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 of removing particles greater than .23 cm (0.09 in.) diameter. A description of the IRWST screen design is provided 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. This analysis must show that the System 80+ screen area 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 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, surfaces in the IRWST that are in direct contact with borated water are lined with stainless steel. IRWST water can be cleaned by the chemical and volume control system. 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 RAI 440.166, the applicant stated that permanent cage-type vortex suppressors, constructed of standard floor grating, will be placed over each ECCS suction inlet. Cage-type vortex suppressors have been found to be effective in suppressing vortices and eliminating air ingestion.

To avoid excessive fouling and plugging of the screens near the IRWST suction inlets during an accident, Section 13.5.2 of CESSAR-DC states in part that the containment must be cleaned of sand, maintenance debris, and other particulate materials prior to startup from a refueling outage.

Several significant events have occurred including the plugging of emergency core cooling system (ECCS) suction strainers at the Perry Nuclear Power Plant and Barsebäck plant in Sweden. The staff had originally proposed that the advanced designs provide the ability for backflushing of the suction strainers, which is similar to the resolution taken in Sweden for the Barsebäck plant. However, a decision was made to evaluate sump sizing criteria, rather than backflushing ability. As a result, in the "Advance Copy of the Final Safety Evaluation Report for the Advanced Boiling Water Reactor (ABWR)," dated December 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 sizing margin.

The staff has reassessed the potential impact of clogging of the ECCS suction strainers on advanced light water reactors. The staff concludes that the System 80+ meets the staff's current 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, Revision 1, for all LOCA scenarios.

System 80+ ?

The staff conducted a qualitative assessment of the risk associated with not applying the three-times multiplier for the ABWR 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 of the LOCA, the staff has decided to take a conservative position. For operating plants, the staff issued NRC Bulletin 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 ~~size~~ 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 reactor designs, as the System 80+ design.

In the DSER, the staff found the resolution of Issue A-43 in conformance with SRP Section 6.2.2, Revision 4 and RG 1.82, Revision 1 and acceptable pending (1) the resolution of the open and confirmatory items in Section 5.4.3 of the DSER, 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, and (2) providing 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 has found 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 acceptable and Issue A-43 is resolved.

Issue A-44: Station Blackout

Issue A-44, formerly USI A-44 in NUREG-0933, is addressed in Section 8.5 of this report. On the basis of its review, the staff concludes that this issue is resolved for the System 80+ design.

basis LOCA had been required by 10 CFR 50.44 well before the 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 accident at Three Mile Island, Unit 2 (TMI-2), 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 post-accident atmosphere that will not support hydrogen combustion. Only those plants whose construction permits had not been issued at the time of the TMI-2 accident are covered by this rule.

In CESSAR-DC Appendix A, ABB-CE originally considered that this is one of the issues applicable to the System 80+ design. Upon further review, in Amendment U, ABB-CE concluded that Issue A-48 was not applicable because the issues had been superceded.

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 now 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 and ABB-CE's response to Issue A-48 is acceptable.

As stated in NUREG-0933, this issue was integrated into the resolution of Issue 121. See also the discussion of Issue 121 in Section 20.2 of this report.

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 is Issue A-49 in NUREG-0933.

(3) "Response to NRC Request for Additional Information," Letter from C.B. Brinkman, ABB-Combustion Engineering, to U.S. Nuclear Regulatory Commission, LD-92-030, February 25, 1992.

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 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 safety-related operator actions (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 ANSI/American Nuclear Society (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.

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In the revised OER (SSAR Amendment Q), ABB-CE indicated 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 noted that the System 80+ design has eliminated the switchover function. 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. The staff finds the information provided by ABB-CE acceptable and, therefore, this issue is resolved.

The DSER stated that this issue would be addressed in the FSER 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: <u>Structural Integrity of Containment Penetrations</u>

Issue B-26 in NUREG-0933 addresses the concern over the staff evaluation to assess the adequacy of specific containment penetration designs from the point of view of structural integrity, in-service inspection (ISI) requirements, and new surveillance or analysis methods applicable to containment penetrations that are identified as inaccessible. However, after reevaluating of the 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 concluded that the issue was resolved and no new requirements were established.

dropped.

It seems mis-leading to say "resolved" when it was really dropped. (I don't think we took any action, which would merit "resolved".)

In CESSAR-DC Appendix A, ABB-CE stated that this is one of the issues considered to be applicable to the design of ALWR. However, after further review, ABB-CE eliminated the issue and categorized it as not relevant to the System 80+ standard design based on the staff's evaluation in NUREG-0933 which concluded that the issue was resolved with no new requirements established. ABB-CE's disposition of this issue is acceptable; therefore, Issue B-26 is resolved.

Issue B-36: <u>Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems</u>

The purpose of B-36 in NUREG-0933 was to develop revisions to the then current guidance and staff technical positions regarding engineered safety feature (ESF) and normal ventilation system air filtration and adsorption units. The

September 1988, Issues A-3, A-4, and A-5, which addressed steam generator 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." The staff concluded that no new or revised requirements were necessary for Issue 66 since NUREG-0844 addressed the safety concerns identified under this issue. Thus, this issue was resolved and no new requirements were established.

Reference A-690 for increased resistance to corrosion.

ABB-CE stated in CESSAR-DC Appendix A that the secondary system, including the steam generators and condenser, will be designed, manufactured, tested, inspected, and operated in accordance with accepted industry codes and standards. The steam generators will meet the requirements of Sections III and XI of the ASME Code for design, manufacture, test, and inspection. Also, steam generator design will meet the intent of the guidance given in SRP Sections 5.4.2.1 and 5.4.2.2 for steam generator materials, quality assurance, inservice tube inspection, and secondary-side water chemistry.

ABB-CE's statements are adequate in ensuring the structural integrity of steam generator tubes and, thus, acceptable in resolving Issue 66 for the System 80+ design. Additional evaluations of steam generator issues are contained in the discussions of Issues 135 and A-4 in this report.

Issue 67.3.3: Improved Accident Monitoring

Issue 67.3.3 in NUREG-0933 addresses weaknesses in reactor system monitoring that could inhibit correct operator responses to events similar to the steam generator tube rupture 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 steam generator 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, 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.

on non-safety-grade components to mitigate design-basis accidents. Issue 70 in NUREG-0933 addressed the assessment of the need for improving the reliability of PORVs and block valves.

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 did not include PORVs and block valves. Instead, the System 80+ includes a safety depressurization system (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.

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ABB-CE did not consider this issue relevant to the System 80+ design because the design does not include PORVs and block valves. In accordance with NUREG-0993, the issue is relevant because it is for all pressurized water reactors which includes the System 80+ design; however, the issue is resolved for the System 80+ design because the design has ~~no PORVs and block valves~~. safety grade valves.

Issue 75: Generic Implications of ATWS Events at the Salem Nuclear Plant

The purpose of Issue 75 in NUREG-0933 was to address 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 reactor protection system. The requirements for this issue were issued in GL 83-28.

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 assure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

of GL 89-10 on safety-related motor-operated valve testing. As stated in NUREG-0933, the related concerns about significant blowdown loads on valves will be addressed in Issue 152.

Section 6.3 of this report discusses the safety injection system for the System 80+ design. Steam lines are not used to power the four high-pressure safety injection pumps, and there are no low-pressure injection pumps. The lack of HPCI steam lines resolves this issue for the System 80+ design.

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ABB-CE did not consider this issue relevant to the System 80+ design because the design does not include ~~steam lines for the HCPI pumps.~~ In accordance with NUREG-0993, the issue is relevant because it is for all pressurized water reactors which includes the System 80+ design; however, the issue is resolved for the Systems 80+ design because the design does not use steam lines for high-pressure safety injection.

Issue 93: <u>Steam Binding of Auxiliary Feedwater Pumps</u>

Issue 93 in NUREG-0933 addressed the potential for a common-mode failure of the auxiliary or emergency feedwater (EFW) system resulting from steam binding of the EFW pumps caused by heated main feedwater leaking back through check valves. The EFW system is used to supply water to the steam generators should the main feedwater (MFW) system be lost, and steam binding of the EFW pumps could result in the loss of the EFW system.

The EFW system 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 main feedwater from entering the EFW system. However, operating experience has shown that check valves tend to leak, thus, permitting hot main feedwater to enter the EFW system. 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)

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, which is consistent with the requirement of the EPRI ALWR utility requirement document (URD) (Ref. 2), is acceptable to the staff 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 structures, systems, and components (SSCs). This approach is acceptable since any sites with a flood level higher than 0.3 m (1 ft) below grade will be excluded from the design certification.

The COL applicant must use the site-specific environmental data for determining PMP in accordance with the guidance of SRP Sections 2.4.2 and 2.4.3 as presented in CESSAR-DC Table 2.0-1, shall not be exceeded by the site-specific flood level. This is a Site Parameter and, therefore, does not require a COL action item as previously referenced in the DSER.

Therefore, based on the above, Issue 103 is resolved for the System 80+ design.

References

(1) "System 80+ Standard Design - CESSAR Design Certification," ABB Combustion Engineering, Windsor, Connecticut, Amendment I, December 21, 1990.

(2) "Advanced Light Water Reactor Utility Requirements Document: Volume I: ALWR Policy and Summary of Top-Tier Requirements; Volume II: ALWR Evolutionary Plant; Volume III: ALWR Passive Plant," Electrical Power Research Institute, Palo Alto, California, August 31, 1990.

Issue 105: Interfacing System LOCA (ISLOCA) at LWRs

Issue 105 in NUREG-0939 was limited to pressure isolation valves (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. ~~However, because ABB-CE did not address Issue 96, low-pressure piping systems due to failure of the RCS boundary isolation could~~

I see no need to say we did not address #96; → it gives the appearance of being deficient.

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 draft NUREG-1449 (DSER Open Item 20.2-13). The staff concludes that the shutdown risk evaluation of ABB-CE is acceptable and is consistent with the formal publication of NUREG-1449.

Therefore, based on the above evaluation, ABB-CE's resolution of Issue 99 is acceptable for the System 80+ design, and DSER Open Item 20.2-13 is resolved. The staff's evaluation of the shutdown risk report is included in Section 19.8A of this report.

Issue 103: Design for Probable Maximum Precipitation (PMP)

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 NOAA hydrometeorological reports. The resolution of this issue calls for the staff to provide guidance on an acceptable means to meet the GDC 2 requirement for design bases for floods to reflect consideration of the most severe historical data with sufficient margin for the limited accuracy, quantity, and period of time in which data have been accumulated. An acceptable resolution is to use SRP Sections 2.4.2 and 2.4.3 as the guidance to incorporate the probable maximum precipitation (PMP) procedures and criteria contained in the latest National Weather Service (NWS) publications.

In CESSAR-DC Appendix A (Ref. 1), 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 including flooding, tornado, and hurricane, and meets the intent of SRP Sections 2.4.2 and 2.4.3. Furthermore, Appendix A requires the COL applicant to review historical site-specific environmental data to ensure compliance with the enveloping assumptions of CESSAR-DC Table 2.0-1.

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.

? which App ? B?

In response to RAI Q440-129 through 151, listed in Appendix of this report, regarding the shutdown risk concerns, ABB-CE submitted a report on July 31, 1992. The report covers the following topics:

(1) procedures
(2) technical specifications improvement
(3) midloop operation
(4) loss of decay heat removal
(5) primary/secondary containment capability and source term
(6) rapid boron dilution
(7) fire protection
(8) instrumentation
(9) ECCS recirculation capability
(10) effect of PWR upper internals
(11) fuel handling and heavy loads
(12) potential for draining the reactor vessel
(13) CESSAR-DC Chapter 15: Non-LOCA/LOCA Dose
(14) CESSAR-DC Chapter 6: Loss-of-Coolant Accidents
(15) CESSAR-DC containment analysis
(16) probabilistic risk assessment

OK

Recently, the staff published a ~~draft~~ report: NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation for Nuclear Plants in the U.S." This

Add to end of first ¶.

result in the rupture of the low-pressure piping and a LOCA, because there is not isolation of the RCS, and, if there are failures in the ECCS, a core melt accident.

As stated in NUREG-0933, Information Notice 92-36 was issued 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.

OK, but do you want to reference SECY-93-087 to be more current? This shows up several times in this writeup.

The staff position regarding intersystem LOCA 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 must justify why it is not practicable to reduce the pressure challenge any further, and also provide compensation 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 pressure isolation valves, (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.

Remember, that proprietary report designation was a mistake on our part. It no longer exists in our files.

Responding to RAI Q440.45, which is listed in Appendix B of this report, Issue 105, ABB-CE submitted its evaluation of various interfacing systems (i.e., chemical and volume control system (CVCS), process sampling system (PSS), seal injection (SI)), seal bleedoff, safety injection system (SIS), and shutdown cooling system (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 that no technical basis was offered to justify why further reducing pressure challenge is not practical. 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. In response, ABB-CE submitted a report on June 15, 1992 (Appendix A of this report, ~~Report CE NPSD-744-P~~) on design features which 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~~ complete its evaluation ~~[illegible]~~ in this Final Safety Evaluation Report.

In Amendment Q of CESSAR-DC, ABB-CE submitted Appendix 5E, "Evaluation of the System 80+ Standard Design to Interfacing System LOCA Challenges," which supersedes the June 15, 1992, report. Appendix 5E provided the ABB-CE evaluation of plant 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. An evaluation was also made on the specific components, such as flanges, valves, pump seals, heat exchangers, vents and drains. This closes out DSER Open Item 20.2-14.

The systems identified to be directly connected to the RCS during some modes of operation are the SCS, SIS, CVCS, and PSS. Each of these systems and associated subsystems is evaluated for compliance with the ISLOCA criteria below.

snubbers 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 Issue A-13 in 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 the staff finds ABB-CE's resolution of Issue 113 for the System 80+ design acceptable.

Issue 118: Tendon Anchorage Failure

Issue 118 in NUREG-0939 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 structural integrity had never been lost. However, the failure of anchor heads to carry the tendon forces could have jeopardized the containment structural integrity during an accident. RGs 1.35 (Rev. 3) and 1.35.1 resolved this issue.

Originally, ABB-CE ~~did~~ not address this issue in Appendix A of CESSAR-DC. ABB-CE did not consider this issue relevant to the System 80+ design because the resolution of the issue did not result in new requirements; however, in accordance with NUREG-0993, the issue is relevant because new requirements (i.e., RGs 1.35 (Rev. 3) and 1.35.1) were established for all light water reactors, which includes the System 80+ design. Subsequently, ABB-CE agreed that the issue could be considered applicable and revised CESSAR-DC accordingly. In CESSAR-DC Section 6.2, ABB-CE stated 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 120: On-Line Testability of Protection Systems

Issue 120 in NUREG-0933 addressed requirements for at-power testing of safety system components without adversely affecting 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 stated in Appendix A of CESSAR-DC that, for the resolution of this issue, the System 80+ design has all-digital instrumentation and control systems, described in Chapter 7 of CESSAR-DC, that allow on-line testing of the systems. The System 80+ RPS and ESFAS, which are discussed 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, this issue is resolved for the System 80+ design.

Issue 121: Hydrogen Control for Large, Dry PWR Containments

Issue 121 in NUREG-0933 was to document the staff's research on hydrogen control in large, dry PWR containments. In response to the TMI-2 accident, the Commission promulgated 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, plants covered by 50.34(f) included only those whose construction permits had not been issued at the time of the TMI-2 accident.

and SECY-93-087

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 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 identified 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 strategies including feed and bleed.

ABB-CE addressed this issue in Appendix A of CESSAR-DC. 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 emergency feedwater flow, reactor coolant temperature, pressure and degree of subcooling, and steam generator pressure and level (wide range).

The safety depressurization system (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 to TMI Task Action Plan Item I.C.1 regarding criteria for feed-and-bleed initiation. In review of the resolutions to Issue I.C.1, the staff found that the current EPGs included in CEN-152 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 plant owner to prepare the plant-specific operating procedures by using feed and bleed to mitigate an accident. Also, in response to RAI Q440.23 regarding acceptability of CEN-152 to the System 80+ design, ABB-CE committed to include the design enhancements, including the SDS in the updated EPGs. Because (1) ABB-CE provided adequate guidelines for mitigation of the feed-and-bleed operation in its current EPGs, (2) ABB-CE ~~committed to~~ updated EPGs to include the SDS design for the System 80+ plant, and (3) the review of the updated EPGs is covered by TMI Task Action Plan Item I.C.1, the resolutions of ABB-CE are acceptable.

AFW from a steam generator affected by a main steamline or feedwater-line break may tend to increase the risk that adequate decay heat removal is not available rather than to decrease it.

ABB-CE stated in Appendix A of CESSAR-DC that the System 80+ emergency feedwater (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 steam generator. Each train consists of

(1) one emergency feedwater storage tank (EFWST)

(2) one 100-percent-capacity motor-driven pump subtrain and one 100-percent-capacity steam-driven pump

(3) flow control valve

(4) isolation valve

(5) check valve l. case

(6) a cavitating ~~V~~enturi

(7) specified instrumentation

One design feature of the EFW system which 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 steam generator 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 EFW system design, the unavailability for the system was estimated from [illegible] studies to be in the range of 1E-04

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ABB-CE stated in Appendix A of CESSAR-DC that the System 80+ emergency feedwater (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 steam generator. Each train consists of

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Because of the improved reliability of the EFW system design, the unavailability for the system was estimated from PRA studies to be in the range of 1E-04

Chapter 19,

to 1E-05 per demand as described in CESSAR-DC Section 10.4.9.1.2. Analysis identified in CESSAR-DC ~~Appendix 10A~~, 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 EFW system meets the recommended unavailability goal of 1E-04 per demand identified in SRP Section 10.4.9 (Rev. 2).

The DSER 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. Based on resolution of those items and this evaluation, the staff finds that ABB-CE's resolution of Issue 124 for the System 80+ design is acceptable.

Issue 125.I.3: <u>Safety Parameters Display System Availability</u>

Issue 125 in NUREG-0933 addressed the long-term actions that came from the issues raised in 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.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, NUREG-0737, required the installation of a SPDS. The purpose of the SPDS is to improve how information is provided to the control room operators by supplying them with continuous information from which the plant safety status can be readily and reliably assessed.

ABB-CE addressed the resolution of Issue 125.I.3 in Appendix A of CESSAR-DC. See Section 18.3.3.2.5 of this report for the staff's evaluation and conclusion that this issue is acceptably resolved. The DSER stated that this issue would be discussed in the FSER 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.

ABB-CE described the resolution of Issue 125.II.7 in Appendix A of CESSAR-DC. The System 80+ design does not include automatic SG isolation logic. The plant design incorporates an emergency feedwater (EFW) system, the designation for AFW in the design, which provides an independent safety-related means of supplying quality feedwater to the steam generator(s) 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 EFW system is a dedicated safety-related system which has no functions for normal plant operation (see CESSAR-DC Section 10.4.9).

The EFW system is designed to be automatically or manually initiated, supplying feedwater to the steam generators for any event that results in the loss of normal feedwater and requires heat removal through the steam generators, 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 to be achieved without actuating the main steam isolation signal.

each of

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 EFW system includes a design requirement that the EFW flow to each SG be restricted by a cavitating Venturi to protect the EFW pump from damage caused by excessive runout flow. The EFW storage has a capacity of 1.32E + 06 L (350,000 gallons) from 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, steam generator overfill, and containment overpressurization.

No need to pick on I&C; SECY-90-377 says its Ok to not complete design detail. Why make it seem we are deficient.

The System 80+ design is a plant that uses digital systems. In CESSAR-DC Section 7.1.1.7, ABB-CE stated that the reactor protection system and engineered safety features 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 is the question of leakage through any isolators used in instrumentation circuits.

In Amendment V,

ABB-CE did not consider this issue relevant to the System 80+ design because it stated that this issue was prioritized as "dropped" or "low" or has not been prioritized. In accordance with NUREG-0993, the issue is relevant because it is prioritized "medium-safety" and is for pressurized water reactors which includes the System 80+ design. ABB-CE ~~reconsidered~~ submitted a writeup summarizing its resolution of this issue.

As discussed in Sections 7.1 and 7.2 of this report, the staff developed the necessary acceptance criteria for digital systems using applicable international and national standards. ~~Although ABB-CE did not complete the hardware and software design for the System 80+ digital I&C systems. Therefore~~, 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," to reach its safety finding for 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 reactor protection system, 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.

Therefore, based on the above, Issue 142 is resolved for the System 80+ design.

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 inservice inspection and testing requirements. Therefore, the staff concludes that the system design meets the applicable acceptance criteria of SRP Section 9.2.2.

The normal chilled water system (NCWS) consists of two equally sized divisions. Each division is sized to provide 100 percent of the cooling capacity required to meet system demands during normal conditions. The NCWS system is not safety related because it is not required to ensure the RCS pressure boundary capability to achieve and maintain safe shutdown, and the ability 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 system 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, and Issue 143 is resolved for the System 80+ design.

Issue 153: <u>Loss of Essential Service Water in LWRs</u>

Issue 153 in NUREG-0933 addressed the reliability of essential service water (ESW) systems and related problems which have been an ongoing staff concern which has been documented in NUREG/CR-2797, IE Bulletins 80-24 and 81-03, GL 89-13, and Issues 51, 65, and 130. In a comprehensive NRC review and evaluation of operating experience related to service water systems, NUREG-1275, Volume 3, a total of 980 operational events involving the ESW system were identified, of which 12 resulted in complete loss of the ESW system. The causes of failure and degradation included: (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) personal and procedural errors.

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 (CDF). 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 resolve by implementation of the requirements stated in GL 89-13.

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 cooling water system). During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to fire protection systems, 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 consider this issue relevant to the System 80+ design because in NUREG-1197 the issue is considered not relevant for design certification. The staff does not use NUREG-1197 to decide whether or not an issue is relevant to a design in accordance with 10 CFR 52.47 for design certification and, in accordance with NUREG-0993, the issue is relevant because the resolution of the issue has resulted in new regulatory requirements and it is for all pressurized water reactors which includes the System 80+ design.

This issue is, however, considered by the staff to be beyond the scope of design certification. The COL applicant will have responsibility for addressing this issue as part of the licensing process. This is COL Action Item 20.2-11. The resolution of this issue is acceptable.

Issue I.A.4.2: <u>Long Term Training Simulator Upgrade</u>

Section 50.34(f)(2)(i) and 50.34(f)(5)(vii) is Issue I.A.4.2 in NUREG-0933 on simulator capabilities. The purpose of this issue was to upgrade the capabilities of the training simulators. This issue was resolved by the publication of Revision 1 to RG 1.149, 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 concludes that the information in this area is outside the scope of the System 80+ standard plant design. The COL applicant ~~referencing the~~ System 80+ certified design will be required to provide site-specific information at the COL phase described in 10 CFR 52.79(b). This is part of COL Action Item 13.2-1. This issue is satisfactorily resolved.

ABB-CE did not consider this issue relevant to the System 80+ design because this issue is an operational issue and not relevant to design certification. In accordance with NUREG-0993, the issue is relevant because the resolution of the issue has resulted in new regulatory requirements and it is for all pressurized water reactors which includes the System 80+ design; however, it

The logic here fails, given the above statement is resolved for the System 80+ design because it is an operational issue. Therefore, ABB-CE's conclusion of relevancy for the System 80+ design is acceptable.

Issue I.C.1: Guidance for Evaluation and Development of Procedures for Transients and Accidents

Issue I.C.1 of NUREG-0933 requires that licensees prepare EOPs. ABB-CE designates these EPGs. The information on EPGs 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 EPGs 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.

To address the concerns in Issue I.C.1, ABB-CE stated in Appendix A of CESSAR-DC that the ultimate responsibility of preparing EPGs to be consistent with guidance in NUREG-0737 and its Supplement 1 remains with the utility owner-operator. However, ABB-CE will assist the owner-operator in preparing EPGs and in training plant operators by providing EPGs as described in applicant's report CEN-152.

The EPGs 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 RAI Q440.23, listed in Appendix B of this report, regarding applicability of the existing EPGs to the System 80+ design, ABB-CE stated that existing EPGs in CEN-152 are applicable to the System 80+ plant. Modifications will be made to the EPGs to account for System 80+ design enhancements, which include (1) four (instead of two in the existing applicant's plants) high-pressure safety injection pumps, (2) additional emergency feedwater pumps, (3) interchangeability of containment spray and shutdown cooling pumps, (4) in-containment refueling water storage tank, (5) safety depressurization system, (6) cavity flooding system, 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 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 EPGs have been modified to reflect the System 80+ design including the design features such as four SI pumps (instead of two high pressure and two low pressure SI pumps in the existing plants of ABB-CE), additional emergency feedwater pumps, interchangebility of containment spray and shutdown cooling pumps, in-containment refueling water storage tanks, alternate ac power supply and safety depressurization system.

(3) The EPGs adequately incorporate the procedure guidelines required for closure of the open items. The EPGs changes for closure or open items are:

of

(a) SI flow rate at the low pressure range see Section 6.3.1 of this report for the closure of Open Item 6.3.1-1.

(b) Use of the RCGVS for RCS pressure control see Section 6.7.1 of this report for the closure of 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 closure of Open Item 6.7.2-4.

(d) Procedures changes reducing challenge to the primary safety valves to open during a SGTR event see Section 15.3.9 of this report for closure of Open Item 15.3.8-1.

(e) Avoidance of deboration during a SGTR event see Section 15.3.9 of this report for the closure of Open Item 15.3.8-2.

Issue I.D.1: Control Room Design Reviews

The purpose of Issue I.D.1 in NUREG-0933 was for licensees to perform a detailed review of their control room using human engineering techniques and guidelines to identify and correct design deficiencies. This issue was clarified in NUREG-0737 and NUREG-700, and is considered resolved. See also Issue I.D.4 in this section.

ABB-CE stated in Appendix A to CESSAR-DC that this issue is summarized in the discussion on human factors engineering in Chapter 18 of CESSAR-DC. Chapter 18 of this report evaluates the human factors engineering of the System 80+ design, including the control room. In Section 18.10, the staff concludes that the ABB-CE human factors engineering program is acceptable and provides an acceptable framework for the human factors interfaces of the design. The basic design features of the control room were reviewed and found consistent with human factors standards, guidelines, and principals, and acceptable for use in the control room. All previously identified DSER issues in Chapter 18 have been resolved. Therefore, Issue I.D.1 is resolved for the System 80+ design.

principle

Issue I.D.2: Plant Safety Parameter Display Console

The purpose of Issue I.D.2 in NUREG-0933 was to improve the information provided to control room operators. The requirements for this issue are in Supplement 1 to NUREG-0737. In NUREG-0933, this issue identifies the need for a safety parameter display system (SPDS) that displays a minimum set of parameters which define the safety status of the plant. Paragraph (2)(iv) of 10 CFR 50.34(f) requires a plant SPDS 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.

In Section 18.7.1.8.1 of CESSAR-DC Amendment Q and the revised OER, ABB-CE indicated how the System 80+ design complied with the SPDS criteria. The staff has reviewed the System 80+ advanced control room design against the

Issue I.F.2: <u>Develop More Detailed Quality Assurance Criteria</u>

The purpose of Issue I.F.2 in NUREG-0933 was to improve the QA program for the design, construction, and operations at 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 quality assurance.

ABB-CE stated in Amendment U of CESSAR-DC that the QA program for the System 80+ design was approved during the NRC staff's review of Section 17.1 of CESSAR-DC. The staff's evaluation of this program is in Section 17.1 of this report. ABB-DC addressed the QA program for the design of System 80+. 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. This is COL Action Item 20.3-3. This satisfactorily resolves this issue for the System 80+ design.

construction

Issue I.G.2: <u>Scope of Test Program</u>

The purpose of Issue I.G.2 in NUREG-0933 was for licensees to develop a more comprehensive preoperational and low-power test program for their plant to find any anomalies in a plant's response to a transient during the initial test program (ITP). With the revisions to the SRP and the then NRC Office of Inspection and Enforcement Manual (June 1989 revision to NUREG-0933), this issue was considered resolved.

Section 14.2 of this report evaluates the initial test program for the System 80+ design. 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 SRP Section 14.2 Phase I through Phase IV test procedures; the COL applicant has the responsibility for the preparation and organization approval of these procedures. These are COL Action Items 14.2.3-1 through 14.2.3-4.

(Based on) ?

The conclusion of Section 14.2 of this report, the ITP for the System 80+ design, is acceptable. This satisfactorily resolves this issue 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 reactor coolant system (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 requirements of 10 CFR Part 50, Appendix A, and meet the applicable codes and standards for the RCS pressure boundary.

ABB-CE stated in Appendix A of CESSAR-DC that the System 80+ design includes a safety depressurization system (SDS) that performs the reactor coolant gas vent (RCGV) function to meet the requirement of TMI Task Action Plan Item II.B.1. The RCGV system is described in CESSAR-DC Section 6.7.1.2.2. The staff has reviewed the design of the RCGV system and concludes, in Section 6.7.1 of this report, that the RCGV system is acceptable because the RCGV system design 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 loss-of-offsite power (GDC 17), and (4) the system must be able to withstand an operating-basis earthquake (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

The purpose of Issue II.B.2 in NUREG-0933 was to have 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 design was to identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. The requirements were issued in NUREG-0737 and the issue is resolved.

ABB-CE stated in Appendix A of CESSAR-DC that a radiation and shielding design review of the System 80+ plant will be performed during the detailed design phase of the plant and referred to 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 accepted by the staff.

The detailed design review of the plant is the responsibility of the COL applicant. Therefore, the completion of this review and the submittal of the review to the staff ~~is COL Action Item 20.3-X~~. is part of the shielding analysis ITAAC.

Therefore, Issue II.B.2 is resolved for the System 80+ design.

Issue II.B.3: Postaccident Sampling Capability

The purpose of Issue II.B.3 in NUREG-0933 was to upgrade post-accident sampling at plants. The requirements are in NUREG-0737 (Rev. 1). The reactor coolant and containment atmosphere sampling-line systems should permit personnel to promptly and safely take a sample under accident conditions. The radiological spectrum analysis facilities should be capable of promptly in their response to SECY-93-087

quantifying certain radionuclides that are indicators of the degree of core damage. In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions.

The staff is currently seeking Commission approval of alternate requirements for 4 of the 11 clarifications in NUREG-0737 on this issue for design certification (Ref. 2).

to N-0737

ABB-CE indicated in Appendix A of CESSAR-DC that System 80+ design includes a process sampling system which permits sampling during reactor operation, cooldown, and postaccident conditions without requiring access to the containment. The staff's evaluation is provided in Section 9.3.2 of this report. As discussed in this section, the Commission approved exemptions so that the capabilities of the post-accident sampling system for the design does not include the determination of the hydrogen concentration in the containment atmosphere and has the time limit for analysis of the reactor coolant boron and radioactivity concentration of 8 and 24 hours, respectively. The conclusion is that the proposal of ABB-CE is adequate in addressing postaccident sampling and, thus, is acceptable in resolving Issue II.B.3 for the System 80+ design.

or chlorides in the reactor coolant

References

(1) U.S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

(2) Enclosure in a letter from D.M. Crutchfield of NRC to E.E. Kintner of GPU Nuclear Corporation, Parsippany, N.J., dated February 27, 1992.

Issue II.B.8: Rulemaking Proceedings on Degraded Core Accidents Description

The purpose of Issue II.B.8 in NUREG-0933 was to consider degraded core accidents in safety reviews of the plant. The work on this issue resulted in a hydrogen control rule that was approved by the Commission and published in the Federal Register on January 25, 1985. With the issuance of this rule on

Issue II.J.3.1: Organization and Staffing to Oversee Design and Construction

The purpose of Issue II.J.3.1 in NUREG-0933 was to require "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."

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 quality assurance and reliability assurance programs for the design, procurement, and fabrication of the System 80+ plant are evaluated in Sections 17.1 and 17.3, respectively, of this report. An inspection of ABB-CE's quality assurance program for design in February 1994 is discussed in Section 17.1 of this report. The conclusion of the staff is that the fundamental requirements for an acceptable design QA program are in place.

The plant organization for the plant design beyond System 80+, the construction of the plant, and modification of the plant is beyond the scope of design certification. The COL applicant will have the responsibility for addressing this issue as part of the COL licensing process. This is identified as COL Action Item 20.2-12. OK if combined with

Based on the above, Issue II.J.3.1 is resolved for the System 80+ design.

In Appendix A of CESSAR-DC, ABB-CE stated that Issue II.J.3.1 was not relevant to the System 80+ design because it had been superceded by other issues. NUREG-0933 did state that this issue is included in Issue I.B.1.1, on organization and management long-term improvements; however, ABB-CE also considered

Issue I.B.1.1 not relevant to the System 80+ design and did not address the issue. Issue II.J.3.1 is considered relevant to the design by the staff in accordance with 52.47(a)(1)(ii) and 50.34(f).

Issue II.J.4.1: Revise Deficiency Reporting Requirements

The purpose of Issue II.J.4.1 in NUREG-0933 was to assure that all reportable items are reported promptly and that the information submitted is complete. The issue was resolved when new requirements were issued in 10 CFR Part 21 and 10 CFR 50.55(e) on July 31, 1991 (56 FR 36081).

The staff evaluated ABB-CE's Section 13.5 of the CESSAR-DC on plant procedures in Section 13.5 of this report. The staff has evaluated 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 satisfactorily resolved for this design in this report.

The plant procedures for adequately reporting in accordance with 10 CFR Part 21 and 10 CFR 50.55(e) is beyond the scope of 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 COL Action Item 20.2-13.

~~In Appendix A of CESSAR-DC, ABB-CE stated that Issue II.J.4.1 was 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.~~ Based on the above, ~~ABB-CE's~~ the resolution of Issue II.J.4.1 for the System 80+ design is acceptable.

The logic is not correct: If it's not relevant, then by definition, there's no resolution by ABB-CE

Issue II.K.3(5): <u>Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident</u>

Issue II.K.3(5) in NUREG-0933 required PWR licensees to study the need for an automatic trip of the reactor coolant pumps and to modify procedures, or design, as appropriate. Licensees should determine how to operate the reactor coolant pump in order to mitigate transients and accidents. Preservation of the maximum reactor coolant system inventory should be considered in the small-break LOCA mitigation; the most effective strategy for decay heat removal should be considered in the other transients' mitigation.

ABB-CE proposed in Appendix A of CESSAR-DC that the RCP operating strategy described in Topical Report CEN-268 for the System 80+ design. CEN-268 justifies the use of the trip two/leave two manual reactor coolant pump trip strategy during transients at 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 topical report was previously reviewed and accepted by the staff (Ref. 1) for implementation of the RCP trip strategy into CEN-152 (CE emergency procedures guidelines). Therefore, this issue is resolved for the System 80+ design.

<u>Reference</u>

(1) Memorandum [from NRC to NRC] from W. Hodges (NRC) to B. Boger, "CEOG Report CEN-268 Revision 1 and CEN-268 Supplement 1, Revision 1," March 3, 1989.

Issue II.K.3(6): <u>Instrumentation To Verify Natural Circulation</u>

Issue II.K.3(6) in NUREG-0933 required licensees to provide instrumentation to verify natural circulation during transient conditions. In accordance with NUREG-0933, the staff determined that this issue was covered by Issues I.C.1, II.F.2, and II.F.3; however, ABB-CE stated in Appendix A of CESSAR-DC that Issue II.K.3(6) was covered by only Issues II.F.2 and II.F.3. Issues I.C.1, II.F.2, and II.F.3 are addressed in this section of this report and the resolutions of these issues for the system 80+ design are acceptable.

u. case

Issue II.K.3(8): <u>Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs</u>

Issue II.K.3(8) in NUREG-0933 addressed further staff consideration of the need for diverse decay heat removal methods independent of steam generators. As stated in NUREG-0933, this issue was covered by Issues I.C.1 and A-45. ABB-CE stated in Appendix A of CESSAR-DC that Issue II.K.3(8) was covered by only Issue A-45.

The staff reviewed the resolutions of Issues I.C.1 and A-45 for the System 80+ design and included its evaluations in this section of this report. Because these evaluations are acceptable, this issue is resolved for the System 80+ design.

Issue II.K.3(25): <u>Effect of Loss of AC Power on Pump Seal</u>

Issue II.K.3(25) required that BWR licensees determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump (RCP) seal coolers. Adequacy of the seal design to withstand a loss-of-offsite power should be demonstrated. This position should prevent excessive loss of reactor coolant system inventory following an anticipated operational occurrence.

normal and CCW

ABB-CE stated in Appendix A of CESSAR-DC that the RCP seals are normally cooled by redundant systems: seal injection from the chemical and volume control system and component cooling water (CCW). In the event of loss of offsite ac power, seal injection can be restored by manually aligning Class 1E power to the charging pump or by using the positive displacement dedicated seal injection pump. Two of the four CCW pumps can be powered from the emergency diesel generators to provide seal cooling.

During a complete loss of ac power (i.e., loss of offsite power and the diesel generators) power can be supplied to the dedicated seal inject on pump, one charging pump, and one CCW pump from the onsite ac power source described in

CESSAR-DC Section 8.3.1.1.5. ABB-CE stated 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 loss of offsite power for at least 2 hours. If seal failure is the consequences of loss of cooling water for 2 hours, an acceptable solution would be emergency power to the CCW 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 and included its evaluation in Section 20.2 of this report.

Therefore, this issue is resolved for the System 80+ design.

Issue II.K.3(30): Revise Small-Break LOCA Methods To Show Compliance With 10 CFR Part 50, Appendix K

Issue II.K.3(30) in NUREG-0933 required 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 semi-scale test facilities. Alternatively, licensees should provide additional justification of the acceptability of present small-break LOCA models with LOFT and semiscale test data.

ABB-CE stated in Appendix A of CESSAR-DC that Topical 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 approval from ABB-CE of small-break LOCA evaluation models. The staff previously evaluated and approved (Refs. 1 and 2) the topical report and concluded that the currently approved small-break LOCA evaluation models are conservative compared with the LOFT and semiscale test data and that they are acceptable for continued use in licensing applications. Therefore, this issue is resolved for the System 80+ design.

determined that this issue was addressed by Issues I.C.1, I.D.2, and I.D.3. ABB-CE stated in Appendix A of CESSAR-DC, that Issue II.K.3(55) was covered by Issue I.D.2.

The resolution of Issues I.C.1, I.D.2, and I.D.3 for the System 80+ design are discussed in this section of this report. The resolutions for these issues are acceptable for the System 80+ design; therefore, the resolution of Issue II.K.3(55) for the System 80+ design is acceptable.

Issue III.A.1.2: Upgrade Licensee Emergency Support Facilities

The purpose of Issue III.A.1.2 in NUREG-0933 was to require 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.

As discussed in Section 13.3 of this report, the System 80+ design provided for a TSC and an OSC. The nearsite EOF is considered by the staff not to be within the System 80+ standard plant design scope and will have to be addressed by the COL applicant referencing the System 80+ standard plant design (COL Action Item 13.3-2). This resolves this issue for the System 80+ design.

Issue III.A.3.3: Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Radio Communication Systems

The purpose of Issue III.A.3.3 in NUREG-0933 was to upgrade the communications at the emergency support facilities at the plant. These communications facilities will be installed by the owner-operator. Therefore, this issue will have to be addressed by the owner-operator who is referencing the System 80+ standard plant design. This is COL Action Item 20.2-12. This resolves this issue for the System 80+ design.

Should not be a COL based on design approval in Chap. 9

20.6 Incorporation of Operating Experience

Background

The NRC staff issues generic communications (bulletins (BLs), generic letters (GLs), and information notices (IN)) to transmit operational experience information to industry. A BL or GL is typically issued when the NRC staff determines that licensees should be required to inform the NRC what actions have or will be taken to address an identified event, condition, or circumstance that is both potentially safety significant and generic. An NRC IN is typically issued when the NRC staff determines that licensees should be informed of an identified 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

1980

ABB-CE has indicated that it has considered operational experience information in the design of the System 80+. In CESSAR-DC Section 1.8, ABB-CE presented the findings of its review of NRC BLs and GLs. ABB-CE determined the applicability of the BL or GL to the System 80+ design, and gave the basis for this determination in CESSAR-DC Tables 1.8-1 and 1.8-2, respectively. As shown in these tables, ABB-CE reviewed the GLs and BLs that were issued by NRC on or after January 1981. This is acceptable to the NRC staff as discussed later in this section, under the ollowing heading: Regulatory Review. These tables close out DSER Confirmatory Item 20.4-1.

ABB-CE also stated in CESSAR-DC Section 1.2 that operational experience information obtained from sources other than NRC BLs and GLs was incorporated into the System 80+ design. In CESSAR-DC Table 1.2-1, ABB-CE described 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 closes out DSER Confirmatory Item 20.4-2.

Regulatory Review

1980

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 upon operating reactor experience), and findings that the staff must make. This document was last revised in April 1982. Significant issues identified before January 1981, were incorporated into this revision. Accordingly, the staff concluded that it is appropriate to focus its review on issues of operational experience identified by NRC since January 1981. As stated above in this section, ABB-CE reviewed and reported on the BLs and GLs issued by the NRC on and after January 1981, as to their applicability to the System 80+ design. ~~Although not requesting ABB-CE to review any earlier issued BLs and GLs, the staff decided to also review the BLs and GLs issued in 1980.~~

As stated above, the NRC BLs and GLs 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 INs do not require a response. Accordingly, the NRC staff concluded that it is appropriate to focus its review on NRC BLs and GLs.

The NRC staff reviewed the NRC BLs and GLs issued since 1980 for incorporation into the NRC staff's System 80+ design review. Upon initial review, BLs and GLs were excluded because they were not relevant to the design of the System 80+ plant, or were associated with TMI action plan items, USIs or GSIs, or existing rules and regulations and, thus, were already an integral part of the NRC 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 20.1 to 20.4 of this chapter. For example, BL 80-01, "Operability of ADS Valve Pneumatic Supply," applies to boiling water reactors, GL 86-14, "Operator Licensing Examinations," relates to operator licensing exam schedules; GL 86-10, "Implementation of Fire Protection Requirements," is associated with 10 CFR 50.48 and/or 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 item, and GL 84-15, "Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability," is associated with a USI/GSI. Additional GLs transmitted previously issued BLs and, therefore, were considered duplicates of the BLs.

The remaining 75 BLs and GLs were reviewed to assure that the issues identified had, if appropriate, been incorporated into the NRC staff's System 80+ design review. Where necessary, additional information was requested 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 was addressed in CESSAR-DC and/or this report. The disposition of the issues identified in the 66 BLs and GLs is summarized in Tables 20.3 and 20.4, respectively, which follows.

Confirm against tables 1.8-3, -4 in Am.

Of the 75 BLs and GLs, 27 issues are being resolved during the ongoing preparation of technical specifications (see Chapter 16 of this report) 9 issues were determined not applicable to the System 80+, and 39 were either not design issues or were already appropriately considered in the System 80+ design.

The staff indicated 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 BLs and GLs 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.

ATTACHMENT 4

B. Starting System

The AAC is provided with redundancy in the starting systems and controls. The system is designed with sufficient capacity for five starts.

C. Cooling System

The AAC is equipped with a self-contained cooling system.

D. Lubrication System

The AAC design includes a pre/post lubrication system that utilizes redundant components.

8.3.1.1.5.4 AAC Periodic Testing

The AAC is designed to be routinely inspected and maintained while the plant is at power.

Instrumentation and controls are provided to permit its synchronization and loading during refueling periods to periodically demonstrate its operability.

Appropriate plant operating procedures shall include periodic testing and/or analysis to verify the adequacy of the AAC to meet the requirements for station blackout and to support its use in Section 3.8 of the Technical Specifications. As a minimum, such procedures shall verify the following:

A. For each Class 1E Division (on an 18 month staggered testing frequency), verify by operating the AAC from the main control room, that the AAC starts within 2 minutes and is capable of energizing the Division's Class 1E buses and supplying all required loads (as defined in the DBA/LOOP LOADS of Tables 8.3.1-2 and 8.3.1-3) within 10 minutes. The steady-state AAC voltage and frequency shall be ≥3744 V and ≤4576 V, and ≥58.8 Hz and ≤61.2 Hz. All AAC starts may be preceded by an engine prelube period.

B. Demonstrate the functionability of all breakers required for the AAC to energize the Class 1E Divisions. This may be performed as part of the above outlined testing, or by separate breaker testing.

C. Each 92 days, verify the AAC starts and achieves steady state voltage (≥3744 V and ≤4576 V), and frequency (≥58.8 Hz and ≤61.2 Hz) within 2 minutes. Load the AAC to ≥90% and ≤100% of its continuous rating and operate it with this load for at least 60 minutes. All AAC starts may be preceded by an engine prelube period.

D The reliability of the AAC is at least 0.95 as calculated by methods defined in NSAC 108, "The Reliability of Emergency Diesel Generators at US Nuclear Power Plant."

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC Ele:trical Power Sources shall be OPERABLE.

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Distribution System; and

b. Two diesel generators (DGs), each capable of supplying one division of the onsite Class 1E AC Distribution System.

c. Automatic load sequencers for Division 1 and Division 2.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One [required] offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for the [required] OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	A.3 Restore [required] offsite circuit to OPERABLE status.	72 hours <u>AND</u> 6 days from discovery of failure to meet LCO

(Continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------------NOTE---------------- Required Action B.3.1 or B.3.2 shall be completed if this Condition is entered. --		
B. One [required] DG inoperable.	B.1 Perform SR 3.8.1.1 for the OPERABLE [required] offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	[24] hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	[24] hours
	AND	
	B.4 Verify the combustion turbine generator (CTG) is functional by verifying the CTG starts and achieves steady state voltage and frequency within [2] minutes.	72 hours
	AND	(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5 Verify the CTG is capable of being aligned to the ESF buses associated with the inoperable DG. AND B.6 Restore [required] DG to OPERABLE status.	72 hours AND Once per 8 hours thereafter 14 days AND 15 days from discovery of failure to meet LCO
C. Two [required] offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one [required] offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features 24 hours
D. One [required] offsite circuit inoperable. AND One [required] DG inoperable.	D.1 ---------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.9. "Distribution Systems - Operating", when Condition D is entered with no AC power source to one division. ------------------------------------ Verify the combustion turbine generator (CTG) is functional by verifying the CTG starts and achieves steady state voltage and frequency within [2] minutes. AND	 12 hours (continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.2 Verify the CTG is capable of being aligned to the ESF buses associated with the inoperabie DG	12 hours AND Once per 8 hours thereafter
	AND	
	D.3 Restore required [offsite] circuits to OPERABLE status.	36 hours
	OR	
	D.4 Restore [required] DG to OPERABLE status.	36 hours
E. Two [required] DGs inoperable.	E.1 Restore one [required] DG to OPERABLE status.	2 hours
F. Required automatic load sequencer inoperable.	F.1 Restore required automatic load sequencer to OPERABLE status.	72 hours
G. Required Actions and associated Completion Times of Conditions A, B, C, D, E, or F not met.	G.1 Be in MODE 3.	6 hours
	AND	
	G.2 Be in MODE 5.	36 hours
H. Three or more [required] AC Power Sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
For the following Surveillances SR 3.0.2 is not applicable: SR 3.8.1.8 through SR 3.8.1.19

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each [required] offsite circuit.	7 days
SR 3.8.1.2	-----NOTES----- 1. Performance of SR 3.8.1.7 satisfies this surveillance. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. [3. A modified DG start, involving idling and gradual acceleration to synchronous speed may be used for the SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances specified in SR 3.8.1.7 must be met.] ----- Verify each DG starts from standby condition and achieves steady state voltage ≥ [3744] volts and ≤ [4576] volts, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.	As specified by Table 3.8.1-1

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load and power factor ranges do not invalidate this test. 3. This surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. -------------------- Verify each diesel generator is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ [5957] kW and ≤ [6255] kW.	As specified by Table 3.8.1-1
SR 3.8.1.4	Verify each day tank [and engine mounted tank] contains ≥ [220] gallons of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank [and engine mounted tank].	31 days
SR 3.8.1.6.	Verify the fuel oil transfer system operates to [automatically] transfer fuel oil from storage tank(s) to the day tank [and engine mounted tank].	92 days
SR 3.8.1.7	----------NOTE---------- All diesel generator starts may be preceded by an engine prelube period. -------------------- Verify each DG starts from standby condition and achieves in ≤ 20 seconds, voltage ≥ [3744] volts and ≤ [4576] volts, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.	184 days

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	------------------------------NOTES------------------------------ 1. This surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events which satisfy this SR. -- Verify automatic and manual transfer of AC power sources from the normal offsite circuit to each alternate [required] offsite circuit.	24 months
SR 3.8.1.9	------------------------------NOTES------------------------------ 1. This surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. -- Verify each DG at a power factor ≤ [0.9] rejects a load of ≥ [1037] kW, and; a. Following load rejection, the frequency is ≤ [63] Hz, b. Within [3] seconds following load rejection, the voltage is ≥ [3744] volts and ≤ [4576] volts; and c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.	24 months
SR 3.8.1.10	------------------------------NOTES------------------------------ 1. This surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. -- Verify each DG operating at a power factor ≤ [0.9] does not trip, and voltage is maintained ≤ [5000] volts during and following a load rejection of ≥ [5957] kW and ≤ [6255] kW.	24 months

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. -------------------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG automatically starts from standby condition and: 1. energizes permanently-connected loads in ≤ 20 seconds, 2. energizes auto-connected shutdown loads through the load sequencer, 3. maintains steady state voltage ≥ [3744] volts and ≤ [4576] volts, 4. maintains steady state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and 5. supplies permanently-connected and auto-connected shutdown loads for ≥ [5] minutes.	24 months

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	---NOTES--- 1. All DG starts may be preceded by an engine prelube period. 2. This surveillance shall not be performed in MODE 1 or 2. 3. Credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated Engineered Safety Features (ESF) actuation signal each DG auto-starts from standby condition and: a. In ≤ 20 seconds after auto-start and during tests, achieves voltage ≥ [3744] V and ≤ [4576] V; b. In ≤ 20 seconds after auto-start and during tests, achieves frequency ≥ [58.8] Hz and ≤ [61.2] Hz; c. Operates for ≥ 5 minutes; d. Permanently-connected loads remain energized from the offsite power system; and e. Emergency loads are auto-connected through the load sequencer to the offsite power system.	24 months

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	----------NOTES---------- 1. This surveillance shall not be performed in MODE 1 or 2. 2. Credit may be taken for unplanned events that satisfy this SR. ------------------------------ Verify each DG automatic trip is bypassed on an [actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal] except: a. Engine Overspeed; b. Generator Differential Current; c. Low Low Lube Oil Pressure; and d. Generator Voltage-Controlled Overcurrent.	24 months
SR 3.8.1.14	----------NOTES---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This surveillance shall not be performed in MODE 1 or 2. 3. Credit may be taken for unplanned events that satisfy this SR. ------------------------------ Verify each DG operating at a power factor ≤ [0.9] operates for ≥ 24 hours: a. For ≥ [2] hours loaded ≥ [6553] kW and ≤ [6881] kW and; b. For the remaining hours of the test loaded ≥ [5957] kW and ≤ [6255] kW.	24 months

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	-----NOTES----- 1. This surveillance shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated in ≥ [2] hours loaded ≥ [5957] kW and ≤ [6255] kW. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. 3. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each DG starts and achieves, in ≤ 20 seconds, voltage ≥ [3744] volts and ≤ [4576] volts, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.	24 months
SR 3.8.1.16	-----NOTES----- 1. This surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite source; and c. Returns to ready to load operation.	24 months

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	---NOTES--- 1. This surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready to load operation; and b. Automatically energizing the emergency loads with offsite power.	[24 months]
SR 3.8.1.18	---NOTES--- 1. This surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. --- Verify the interval between each sequenced load block is within ± 10% of design interval for each emergency and shutdown load sequencer.	[24 months]
SR 3.8.1.19	---NOTES--- 1. All DG starts may be preceded by an engine prelube period. 2. This surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. ---	

(Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19 (continued)	Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG automatically starts from standby condition and: 1. energizes permanently-connected loads in ≤ 20 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage ≥ [3744] volts and ≤ [4576] volts, 4. achieves steady state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and 5. supplies permanently-connected and auto-connected emergency loads for ≥ [5] minutes.	[24 months]
SR 3.8.1.20	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. -- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ 20 seconds, voltage ≥ [3744] volts and ≤ [4576] volts, and frequency ≥ [58.8] Hz and ≥ [61.2] Hz.	10 years

Table 3.8.1-1 (Page 1 of 1)
Diesel Generator Test Schedule

NUMBER OF FAILURES IN LAST 25 VALID TESTS[a]	FREQUENCY
≤ 3	31 days
≥ 4	7 days[b] (but no less than 24 hours)

[a] Criteria for determining number of failures and valid tests shall be in accordance with Regulatory Position C.2.1 of Regulatory Guide 1.9, Revision 3, where the number of tests and failures is determined on a per DG basis.

[b] This test frequency shall be maintained until seven consecutive failure free starts from standby conditions and load and run tests have been performed. If, subsequent to the 7 failure free tests, 1 or more additional failures occur, such that there are again 4 or more failures in the last 25 tests, the testing interval shall again be reduced as noted above and maintained until 7 consecutive failure free tests have been performed.

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC Electrical Power Sources shall be OPERABLE:

a. The qualified circuit(s) between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System required by LCO 3.8.10, "Distribution System - Shutdown"; and

b. On-site power source(s) capable of supplying the division(s) of the onsite Class 1E AC Electrical Power Distribution System required by LCO 3.8.10, "Distribution Systems - Shutdown".

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	-----------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required division de-energized as a result of Condition A. --	
A. Required offsite circuits inoperable.	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend handling of irradiated fuel assemblies.	Immediately

(Continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required offsite circuit(s) inoperable. (continued)	AND	
	A.2.3 Initiate actions to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	
	A.2.5 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. All required on-site power source(s) inoperable.	B.1 Declare affected required feature(s) with no on-site power available INOPERABLE.	Immediately
	OR	
	B.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	B.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	B.2.4 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	(continued)

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	(continued)	B.2.5	Initiate action to restore required DG to OPERABLE status.	Immediately
	------------NOTE----------- Required Action C.3.1 or C.3.2 shall be completed if this condition is entered. -------------------------------			
C.	With one of the two required on-site sources inoperable.	C.1	Perform SR 3.8.1.1 for the OPERABLE [required] offsite circuit(s).	1 hour AND Once per 8 hours thereafter
		AND		
		C.2	Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
		AND		
		C.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	[24] hours
		OR		
		C.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	[24] hours
		AND		
		C.4	Verify the combustion turbine generator (CTG) is functional by verifying the CTG starts and achieves steady state voltage and frequency within [2] minutes.	4 hours
		AND		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.5	Verify the CTG is capable of being aligned to the ESF buses associated with the inoperable D/G	4 hours AND Once per 8 hours thereafter
	AND		
	C.6	Restore [required] DG to OPERABLE status.	14 days
D. Required Actions and Completion Times of Condition C not met.	D.1	Declare affected required feature(s) with no on-site power available INOPERABLE.	Immediately
	OR		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
		AND	
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
		AND	
	D.2.3	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
		AND	
	D.2.4	Initiate action to suspend operations involving positive reactivity additions.	Immediately
		AND	
	D.2.5	Initiate action to restore required DG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.2.1	------------------------------NOTE------------------------------ SR 3.8.1.3 is only required to be performed when more than the minimum number of AC Sources required by LCO 3.8.2 are available, but at least once every 6 months. -- For AC sources required to be OPERABLE, the SRs of LCO 3.8.1, "AC Sources - Operating" are applicable.	In accordance with applicable SRs

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The AC Power Sources consist of the offsite power sources (preferred power) and the onsite standby power sources (Division 1 and Division 2 diesel generators). In addition, a Combustion Turbine Generator (CTG) provides a diverse on-site AC standby power source. As required by 10CFR50, Appendix A General Design Criterion 17 (Ref. 1), the design of the AC power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Division 1 and 2 onsite Class 1E AC Distribution System is divided into redundant load groups (divisions) so that loss of any one group will not prevent the minimum safety functions from being performed. Each division has connections to two preferred (offsite) power supplies and to a single diesel generator.

A qualified circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses. The AC Distribution System consists of four (4) qualified circuits.

Independent transmission lines supply offsite power to Preferred Switchyards I & II. Preferred Switchyard I feeds the Unit Main Transformer (UMT) and Preferred Switchyard II feeds the Reserve Auxiliary Transformers (RATs). The UMT transforms [230 kV] to [24 kV]. This [24 kV] is fed to two Unit Auxiliary Transformers (UATs). These UATs each provide power to their respective separate switchgear groups X and Y.

UATs are the normal preferred source of power to the [4160 volt] emergency buses. X-UAT provides the power to Division 1 emergency buses and Y-UAT provides the power to Division 2 emergency buses. Backup offsite power for either or both the emergency buses is provided through the RATs (1 per division). If offsite power is not available, the emergency buses are supplied from their respective diesel generator, (DG). DG1 supplies power to Division 1 emergency buses and DG2 supplies power to Division 2 emergency buses.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1 minute] after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.

The onsite standby power source for each division ESF bus is a dedicated DG. The DGs start automatically on a Safety Injection Actuation Signal (SIAS) or on a loss of voltage (LOV) on the respective emergency buses. Even though the DGs are started on SIAS,

(continued)

BASES

BACKGROUND (continued)

they will not power the emergency buses unless the offsite sources of power are unavailable. The DG automatically ties to its buses on a LOV condition on that bus with offsite power unavailable.

Following the trip of offsite power, [a sequencer/an undervoltage signal] strips nonpermanent loads from the ESF buses. When the DG is tied to the ESF buses, loads are then sequentially connected to its respective ESF buses by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within [1] minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

In accordance with Regulatory Guide 1.9 (Ref. 2), diesel generators 1 and 2 have [6067] kW continuous and [6674] kW two-hour load ratings. The diesel generators are rated at [4160 volts], three phase, 60 Hz, and are capable of attaining required frequency and voltage within twenty seconds after receipt of a start signal (Ref. 3). The ESF systems which are powered from divisional power sources are listed in Reference 3.

The CTG is a diverse and independent non-Class 1E on-site power source provided for coping with a Station Blackout (SBO) and a Loss of Offsite Power (LOOP) scenarios. The CTG is located within the protected area and it will start automatically, within [2] minutes from the onset of a LOOP event. In addition, the CTG is automatically connected to the de-engergized 4.16 kV Permanent Non-Safety buses. Alignment to the Class 1E ESF buses is accomplished from the control room. The CTG is sized to accommodate one Safety Division loads for a worst case unit shutdown to cold shutdown and/or DBA and one division of Permanent Non-Safety loads.

The CTG is Quality Class 2 and is designed with a High Confidence of Low Probability of Failure (HCLPF) value of .36g. This PRA-Bases Seismic Margin Assessment (SMA) provides assurance that the CTG will be available to back up the DGs for seismic events on the order of the design basis earthquake of .3g. (Ref. 15). This robust design includes the enclosure and the support systems of the CTG.

Other external events which could affect CTG availability as a backup to the DG are hurricanes and tornados. Due to early warning systems the plant will be required to shutdown as a hurricane approaches. For the tornado it is assumed the CTG will not be available.

BASES

BACKGROUND (continued)	A PRA for the CTG's contribution to CDF was performed. The base case assumed a tornado along with all internal events and resulted in a 2.0 E-6 contribution to CDF. With a 14 day unavailability of the DG and the CTG verified to be functional, the CDF increased approximately 4% to 2.08 E-6. This PRA provides an assurance that the CTG can be substituted for the DG without adversely impacting CDF for internal events and tornado strikes.
APPLICABLE SAFETY ANALYSES	The initial conditions of DBA and transient analyses in CESSAR-DC Chapters 6 (Ref. 4) and 15 (Ref. 5) assume ESF systems are OPERABLE. The AC Power System is designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These design limits are discussed in more detail in the Bases for LCO Sections 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems). In general, the safety analysis considered offsite power to be available to ESF equipment following event initiation. Offsite power is not considered to be safety-related. A loss of offsite power (LOOP) alone is an analyzed event since it presents a challenge to the plant's safety features and would result in a total loss of AC power if the diesel generators and the combustion turbine failed to start. The OPERABILITY of an offsite AC source is not explicitly required by the safety analyses. Therefore, the need for two qualified circuits was not derived from the safety analysis, since events postulating failure of offsite power considered a complete loss of offsite power. Such events disable all offsite circuits. The requirement for two qualified circuits was derived from the design criteria (Ref. 1) and standards incorporated into the plant design, which required redundant, independent offsite power sources. The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon maintaining at least one division of the AC and DC Power Sources and associated distribution systems OPERABLE during accident conditions in the event of (1) an assumed loss of all offsite or all onsite AC power, and (2) a worse case single failure. The AC sources satisfy Criterion 3 of the NRC Policy Statement.
LCO	Two qualified circuits (Ref. 3) between the offsite transmission network and the onsite Class 1E AC Distribution System, and the two independent diesel generators (Ref. 3) each capable of supplying one division of the onsite Class 1E AC Distribution System, ensure availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated design basis accident (DBA). Qualified offsite circuits are those that are described in CESSAR-DC and are part of the licensing basis for the unit.