ABB

April 14, 1994
LD-94-024

Docket No. 52-002

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

Subject: System 80+™ Information for Issue Closure

Reference: NRC Letter, R. W. Borchardt to C. B. Brinkman (ABB-CE), dated March 8, 1994

Dear Sirs:

The attachments to this letter provide revisions to CESSAR-DC and comments on the System 80+ FSER. Attachment 1 provides additional structural design detail and revisions to CESSAR-DC discussed with Mr. T. Cheng and Mr. S. Ali. Attachemnt 2 provides our comments on the FSER, per the referenced letter. Attachment 3 provides a clean copy of technical revisions to the Technical Specifications which were agreed upon at the April 5-6, 1994, meeting with NRC staff. These should be given to Mr. M. Reinhart. These revisions and the agreed-upon format changes will be formally printed in Amendment W (May 31, 1994).

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

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)
P. Lang (DOE)

D032
1/1

ABB Combustion Engineering Nuclear Power

Combustion Engineering, Inc.
P.O. Box 500
1000 Prospect Hill Rd.
Windsor, CT 06095
Telephone (203) 688-1911
Fax (203) 285-5203

ATTACHMENT 1

10.5.7.2 Design Loads (Reference Section 3.8.4.3)

A 225 ton overhead bridge crane must be provided over the shipping bay.

Refer to Table 3.8A-1 for additional design loads applicable to the CVCS Area.

11.0 SUPPLEMENTAL DESIGN CRITERIA FOR NON-NUCLEAR ISLAND, SEISMIC CATEGORY I AND II STRUCTURES

11.1 DIESEL FUEL STORAGE STRUCTURE

11.1.1 BUILDING CLASSIFICATION

- Quality Class 1
- Safety Class 3
- Seismic Category I

11.1.2 DESCRIPTION

There are two Diesel Fuel Storage Structures; one on each side of the Nuclear Island.

The main reinforced concrete structure is approximately 12'-6" ~~25~~ ft high, 63 ft long and 44 ft wide founded on a ~~two foot~~ 2'-3" thick reinforced concrete mat located ~~13'-6"~~ below the grade elevation of 90'-9". The walls and the roof are ~~two foot~~ 2'-3" thick. There is a two foot thick center reinforced concrete wall that divides the structure into two separate bays. Each bay encloses a diesel fuel oil tank, a tank vent, a sump with a sump pump, and necessary piping. The bays are separated from each other and from the equipment room by ~~two~~ three-hour rated fire barriers (i.e., 2 ft thick walls). A steel platform at elevation ~~91'-0"~~ 89'-2" surrounds each of the fuel tanks. The outside doors are protected against tornado missiles by a concrete missile barrier.

There is also an attached outside Seismic Category II equipment room that is approximately ~~8~~ 10 ft high, 12 ft long and 28 ft wide founded on a ~~8"~~ 15" reinforced concrete mat. The equipment room is a steel framed structure with insulated metal siding and a metal deck roof.

The Diesel Fuel Storage Structure shall be located a minimum of 50 feet from any hydrogen storage area to preclude loading to the structure from a potential hydrogen burn.

11.1.3 ELEVATIONS

- El. ~~77'-3"~~ 78'-3" Bottom of base mat for the main structure
- El. ~~90'-9"~~ 91'-9" ~~Bottom~~ Top of base mat for the equipment room structure
- El. ~~90'-9"~~ Top of steel platform
- El. ~~102'-3"~~ 103'-3" Top of roof

11.1.4 CODES AND STANDARDS

The codes and standards applicable to Seismic Category I buildings shall be met for the Diesel Fuel Storage Structure including the equipment room.

11.1.5 LOADS

In addition to the minimum design loads requirements of Section 5.1 of this appendix, the following additional specific load requirements shall be met. Should conflicting values occur between this section and Section 5.1 of this appendix, the values specified in this section apply.

11.1.5.1 Dead Load (D)

The foundation slab shall be designed to include the reactions imparted by the steel fuel tank support frames. The weight of each tank and oil is approximately 402 kips. (The site specific SAR shall verify the tank volume is adequate for the diesel generators purchased, such that they meet their design criteria.) The tank support frame is not covered by this criteria and shall be designed in accordance with the rules of Reference ASME Section III, Division I, Subsection NF.

11.1.5.2 Live Load (L)

The Diesel Fuel Storage Structure shall be designed for the following floor live load.

Item	Live Load
• Basemat Floor	250 psf
• Steel Platform	150 psf
• Roof	100 psf

11.1.5.3 Temperature Loads (T_o)

The normal concrete surface operating temperature within the building ranges from 60°F to 90°F. The ambient temperature range outside of the building shall be -10°F to 100°F (See Section 5.1.1.5 of this appendix). Site specific provisions may be taken to minimize the effects of the structural temperature gradient produced by these conditions.

11.1.5.4 Seismic Loads (E')

The seismic accelerations shall be as specified in Table 3.8A-2.

11.1.5.5 Oil Leakage

All building walls shall be designed to contain the contents of the 45,000 gallon oil tanks in the event one tank fails.

11.1.5.6 Other Loads

All abnormal loads (i.e., P_a, T_a, R_a, Y_j, Y_m and Y_r) are zero.

11.1.6 LOADING COMBINATIONS AND ACCEPTANCE CRITERIA

11.1.6.1 Concrete

The requirements of Section 5.2.2 of this appendix shall be met.

Amendment U
December 31, 1993

11.1.6.2 Stability

The requirements of Section 5.2.4 of this appendix shall be met.

11.1.7 OTHER REQUIREMENTS

The building is to be founded on competent structural backfill as defined in Section 10.1 of this appendix. The bearing pressure shall not exceed the allowable value given in Table 2.0-1.

11.2 COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE

11.2.1 BUILDING CLASSIFICATION

- Quality Class 1
- Safety Class 3
- Seismic Category I

11.2.2 DESCRIPTION

There are two Component Cooling Water (CCW) Heat Exchanger Structures, each structure houses two heat exchangers. The CCW system is a redundant system with only two heat exchangers required for plant operation. The first floor houses the heat exchanger, while the basemat levels contains piping and equipment.

Each structure is a two story reinforced concrete structure approximately 34 ft high, from the top of the mat, 110 ft long, and 44 ft wide founded on a four foot thick reinforced concrete mat located ~~17'-6"~~ 17'-0" below grade. The walls are 2'-3" thick and the roof ~~are~~ is two foot thick. The first floor of the structure is three floor thick and is supported by three rows columns approximately twenty two feet on center with the two outer rows located directly under the two heat exchangers. The center row of these columns is continued through the first floor to provide additional support for the roof.

The roof supports two fan rooms on one end of the building and two air inlet rooms on the opposite end of the building. Both of these rooms extend the width of the building and are approximately 23 feet wide with a partially open face covered with a bird screen. A concrete overhang is provided and serves as a missile barrier for the open face.

The outside doors are protected against tornado missiles by concrete missile barriers.

CCW heat exchanger maintenance sumps are located in the basemat at one end of the structure. The sump has a capacity equal to the fluid contents of the shell inside of one heat exchanger. There are floor drain sumps located at the opposite end of the structure.

The CCW Heat Exchanger Structures shall be located a minimum of 50 feet away from any hydrogen storage area to preclude loading to the structure from a potential hydrogen burn.

An underground tunnel is connected to each CCW Heat Exchanger Structure from the Nuclear Annex for the CCW piping. The top of the tunnels basemat is at the same elevation as the top of the CCW Heat Exchanger Structure basemat. IV

11.2.3 ELEVATIONS

- El. ~~120'-9"~~ 121'-9" Top of roof of fan/air filter room
- El. ~~110'-9"~~ 111'-9" Top of Roof
- El. ~~90'-9"~~ 91'-9" Top of the first floor ~~(grade)~~ (1ft above grade)
- El. ~~72'-9"~~ 73'-9" Bottom of basemat

V

11.2.4 CODES AND STANDARDS

The codes and standards applicable to Seismic Category I buildings shall be met.

11.2.5 LOADS

In addition to the minimum design loads requirements of Section 5.1 of this appendix, the following additional specific load requirements shall be met. Should conflicting values occur between this section and Section 5.1 of this appendix, the values specified in this section apply.

11.2.5.1 Dead Load (D)

The weight of each heat exchanger when full of water is approximately 250 Kips excluding the heat exchanger saddle and leg supports. The heat exchanger support is not covered by this criteria and shall be designed in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NF.

11.2.5.2 Live Load (L)

The CCW Heat Exchanger Structure shall be designed for the following live loads.

Item	Live Load
• Fan and Air Inlet Room	150 psf
• Roof	100 psf
• First floor	150 psf
• Basemat	250 psf

11.2.5.3 Temperature Loads (T_o)

The normal concrete surface operating temperature within the building ranges from 60°F to 90°F. The ambient temperature range outside of the building shall be assumed to range from -10°F to 100°F (See Section 5.1.1.5 of this appendix). Site specific provisions may be taken to minimize the effects of the structural temperature gradient produced by these conditions.

11.2.5.4 Seismic Loads (E')

The seismic accelerations shall be as specified in the Table 3.8A-3.

The maximum differential settlements that can be tolerated by the basemat are calculated based on the moment capacity. Any settlements less than those shown are acceptable from the standpoint of stress in the basemat. A settlement monitoring program ensures that proper consideration is given to actual settlements during and after construction.

6.0 TYPICAL ACI-318 CHAPTER 21 DUCTILITY CONNECTION DETAILS

The System 80+ design incorporates ACI-318 Chapter 21 ductility requirements as identified in Appendix 3.8A, Section 6.2.1.1.

Figure 3.8B-2 is provided as a supplement to illustrate the ductility steel requirements in Appendix 3.8A. Typical details are shown with a description and a reference to the ACI-318 Code section associated with the detail. The actual spacing, dimensions, reinforcing bar sizes, bend angles, etc., are obtained from the ACI-318 Code sections. The details provided are for illustration purposes only.

7.0 NON-NUCLEAR ISLAND STRUCTURES

7.1 DIESEL FUEL STORAGE STRUCTURE

7.1.1 DESCRIPTION OF STRUCTURE

The Diesel Fuel Storage Structure is a two bay, partially embedded, single-story reinforced concrete building; symmetrical about its north-south axis. Each bay houses a single diesel fuel oil tank.

The specified concrete compression strength is 4,000 psi and the specified minimum yield strength of the reinforcing steel is 60,000 psi.

7.1.2 ANALYSIS METHODS

The Diesel Fuel Storage Structure is analyzed for the design loads described in Appendix 3.8A to determine the global and localized member forces for which the structure must be designed.

The structure is analyzed using a linear elastic three-dimensional finite element flat plate type model supported on elastic soil springs. Thermal and equivalent static loads corresponding to the various individual loading conditions identified in Sections 3.8A.5.1 and 3.8A.11.1.5 are applied to the structure model and the resulting member forces and moments computed. The resulting member forces are combined in accordance with the load combinations, specified in Section 5.2.2 of Appendix 3.8A, to determine the design loads for the critical sections.

7.1.3 LOADS AND LOAD COMBINATIONS

The Diesel Fuel Storage Structure is evaluated for the loads and load combinations specified in Sections 3.8A.5.1 and 3.8A.5.2, respectively, for Seismic Category I concrete structures.

The major loadings affecting the design of the structure are dead loads (i.e., self weight and equipment weight from the diesel fuel storage tanks), temperature, static and dynamic lateral soil and ground water pressures, wind loads, earthquake loads, and tornado loads.

The critical load combinations are equations 5.2.2.1(a), 5.2.2.1(d), and 5.2.2.2(a) of Section 3.8A.5.2, i.e.,

$$U = 1.4D + 1.7L$$

$$U = 0.75\,(1.4D + 1.7F + 1.7L + 1.7H + 1.7T_o + 1.7R_o)$$

$$U = D + F + L + H + T_o + R_o + E'$$

7.1.4 ANALYSES AND RESULTS

The reinforced concrete members of Seismic Category I structures are designed to the criteria specified in ACI 349 and NRC Regulatory Guide 1.142, except as modified by Appendix 3.8A (see 3.8A.6.2). In general, symmetrical reinforcing steel (i.e., the same area and configuration on opposite faces of members), is provided except in local areas. Concrete joints shall be detailed in accordance with the criteria specified in ACI 318, Chapter 21 (see Section 3.8A.6.2.1.1.1 and Section 6.0 of this appendix).

Foundation Mat:

The primary flexural reinforcing for the ~~two foot~~ 2'-3" thick foundation mat consists of a rectangular grid of #11 at ~~18~~ 12 inches each way/each face, [i.e., ~~[illegible]~~ 1.56 in²/ft].

No transverse shear reinforcing is required.

East and West Walls:

The primary flexural reinforcing for these ~~two foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 8 inches each way/each face, [i.e., 2.34 in²/ft].

No transverse shear reinforcing is required.

North and South Walls:

The primary flexural reinforcing for these ~~two foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches each way/each face, [i.e., 3.12 in²/ft].

Transverse shear reinforcing consisting of #6 at 12 inches is required in both directions for the entire wall area. The shear steel extends 9 ft down from the top of the roof.

Center Wall:

The primary flexural reinforcing for this two-foot thick wall consists of a rectangular grid of #11 at 6 inches each way/each face, [i.e., 3.12 in^2/ft]. Compression ties are required for the top half of the wall.
No transverse shear reinforcing is required.

Roof:

The primary flexural reinforcing for these ~~two-foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches each way/each face [i.e., 3.12 in^2/ft].

No transverse shear reinforcing is required.

7.1.5 CONCLUSION

The concrete and reinforcing steel section strengths of the Diesel Fuel Storage Structure are sufficient to resist the design basis load and load combination criteria specified in Sections 3.8A.11.1 and 3.8A.5.0. Typical reinforcing details are shown in Figures 3.8B-5 and 3.8B-6.

7.2 COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE

7.2.1 DESCRIPTION OF STRUCTURE

The Component Cooling Water Heat Exchanger Structure is a single bay, partially embedded, two-story reinforced concrete building. The top floor houses two heat exchangers supported on saddles which spread the loadings to the supporting floor and column system.

The specified concrete compression strength is 4,000 psi and the specified minimum yield strength of the reinforcing steel is 60,000 psi.

7.2.2 ANALYSIS METHODS

The Component Cooling Water Heat Exchanger Structure is analyzed for the design loads described in Appendix 3.8A to determine the global and localized member forces for which the structure must be designed.

The structure is analyzed using manual computations which consider the structure to be comprised of linear elastic one-way wall and slab panels. Thermal and equivalent static loads corresponding to the various individual loading conditions identified in Sections 3.8A.5.1 and 3.8A.11.2.5 are applied to the one-way panel models and resulting member forces and moments computed. The resulting member forces are combined in accordance with the load combinations, specified in Section 5.2.2 of Appendix 3.8A, to determine the design loads for the critical sections.

7.2.3 LOADS AND LOAD COMBINATIONS

The Component Cooling Water Heat Exchanger Structure is evaluated for the loads and load combinations specified in Sections 3.8A.5.1 and 3.8A.5.2, respectively, for Seismic Category I concrete structures.

The major loadings affecting the design of the structure are dead loads (i.e., self weight and equipment weight from the CCW heat exchangers), temperature, static and dynamic lateral soil and ground water pressures, wind loads, earthquake loads, and tornado loads.

The critical load combinations are equations 5.2.2.1(a), 5.2.2.1(d), and 5.2.2.2(a) of Section 3.8A.5.2, i.e.,

$$U = 1.4D + 1.7L$$

$$U = 0.75\,(1.4D + 1.7F + 1.7L + 1.7H + 1.7T_o + 1.7R_o)$$

$$U = D + F + L + H + T_o + R_o + E'$$

7.2.4 ANALYSES AND RESULTS

The reinforced concrete members of Seismic Category I structures are designed to the criteria specified in ACI 349 and NRC Regulatory Guide 1.142, except as modified by Appendix 3.8A (see 3.8A.6.2). In general, symmetrical reinforcing steel (i.e., the same area and configuration on opposite faces of members), is provided except in local areas. Concrete joints shall be detailed in accordance with the criteria specified in ACI 318, Chapter 21 (see Section 3.8A.6.2.1.1.1 and Section 6.0 of this appendix).

Foundation Mat:

The primary reinforcing for the four-foot thick foundation mat consists of a rectangular grid of #9 at 10 inches ~~each way~~ each face, [i.e., 1.20 in²/ft] in the long direction and #11 at 6 inches each face, [i.e., 3.12 in²/ft] in the short direction.
No transverse shear reinforcing is required.

East and West Walls (Short Direction):

The primary reinforcing for these ~~two-foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches ~~each way~~ each face, [i.e., 3.12 in²/ft] vertically and #11 at 10 inches each face, [i.e., 1.87 in²/ft] horizontally.
No transverse shear reinforcing is required.

North and South Walls (Long Direction):

The primary reinforcing for these ~~four-foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches vertically each face and #11 at 10 inches horizontally, [i.e., 3.12 in²/ft and 1.87 in²/ft, respectively].

No transverse shear reinforcing is required.

Floor Slab at Elevation 90'-9":

The primary reinforcing for the ~~three foot~~ 3'-0" floor slab consists of a rectangular grid of #10 at 10 inches ~~each way/~~each face, [i.e., 1.52 in^2/ft] in the long direction and #11 at 10 inches each face, [i.e., 1.87 in^2/ft] in the short direction.
No transverse shear reinforcing is required.

Roof Slab at Elevation 110'-9":

The primary reinforcing for the ~~two foot~~ 2'-3" thick roof consists of a rectangular grid of #11 at 10 inches each way/each face, [i.e., 1.87 in^2/ft].

No transverse shear reinforcing is required.

7.2.5 CONCLUSION

The concrete and reinforcing steel section strengths of the Component Cooling Water Heat Exchanger Structure are sufficient to resist the design basis load and load combination criteria specified in Sections 3.8A.11.2 and 3.8A.5.0. Typical reinforcing details are shown in Figures 3.8B-7 through 3.8B-9.

7.3 COMPONENT COOLING WATER TUNNEL

7.3.1 DESCRIPTION OF STRUCTURE

The Component Cooling Water Tunnel is a single compartment, fully embedded, one-story reinforced concrete structure. The tunnel houses and protects the Component Cooling Water piping which is routed from the corresponding Nuclear Island pipe chase to the basement of the Component Cooling Water Heat Exchanger Structure. The tunnel is attached at one end to the Nuclear Island Pipe Chase and the Component Cooling Water Heat Exchanger Structure at the other end via flexible connections. The flexible connections allow differential movement between the three structures without transferring loadings.

The specified concrete compression strength is 4,000 psi and the specified minimum yield strength of the reinforcing steel is 60,000 psi.

7.3.2 ANALYSIS METHODS

The Component Cooling Water Tunnel is analyzed for the design loads described in Appendix 3.8A to determine the global and localized member forces for which the structure must be designed.

The structure is analyzed using manual computations which consider the structure to be comprised of linear elastic one-way wall and slab panels. The lateral loads on the tunnel were evaluated using a linear elastic frame model with a unit width. Thermal and equivalent static loads corresponding to the various individual loading conditions identified in Sections 3.8A.5.1 and 3.8A.11.5 are applied to the equivalent frame model and resulting member forces and moments computed. The resulting

.7

member forces are combined in accordance with the load combinations, specified in Section 5.2.2 of Appendix 3.8A, to determine the design loads for the critical sections.

7.3.3 LOADS AND LOAD COMBINATIONS

The Component Cooling Water Tunnel is evaluated for the loads and load combinations specified in Sections 3.8A.5.1 and 3.8A.5.2, respectively, for Seismic Category I concrete structures.

The major loadings affecting the design of the structure are dead loads (i.e., self weight and equipment weight from the piping systems), AASHO H20-44 truck overburden pressure, temperature, static and dynamic lateral soil and ground water pressures, tornado loads, and earthquake loads (including seismic inertia and wave passage). Seismically induced forces due to differential movements are eliminated by providing flexible connections, at each end of the tunnel, which are capable of accommodating the movements without transferring loads.

The critical load combinations are equations 5.2.2.1(a), 5.2.2.1(d), and 5.2.2.2(a) of Section 3.8A.5.2, i.e.,

$$U = 1.4D + 1.7L$$

$$U = 0.75\,(1.4D + 1.7F + 1.7L + 1.7H + 1.7T_o + 1.7R_o)$$

$$U = D + F + L + H + T_o + R_o + E'$$

7.3.4 ANALYSES AND RESULTS

The reinforced concrete members of Seismic Category I structures are designed to the criteria specified in ACI 349 and NRC Regulatory Guide 1.142, except as modified by Appendix 3.8A (see 3.8A.6.2). In general, symmetrical reinforcing steel (i.e., the same area and configuration on opposite faces of members), is provided except in local areas. Concrete joints shall be detailed in accordance with the criteria specified in ACI 318, Chapter 21 (see Section 3.8A.6.2.1.1.1 and Section 6.0 of this appendix).

Foundation Mat:

The primary reinforcing for the three-foot thick foundation mat consists of a rectangular grid of #~~11~~ 9 at ~~10~~ 12 inches each way/each face, [i.e., ~~1.87~~ 1.00 in^2/ft].

No transverse shear reinforcing is required.

~~East and West~~ Walls:

The primary reinforcing for these two-foot thick walls consists of a rectangular grid of #~~11~~ 9 at ~~10~~ 12 inches each way/each face, [i.e., ~~1.87~~ 1.00 in^2/ft].

No transverse shear reinforcing is required.

~~North and South Walls:~~

~~The primary reinforcing for these two-foot thick walls consists of a rectangular grid of #11 at 10 inches each way/each face, [i.e., 1.87 in^2/ft].~~

~~No transverse shear reinforcing is required.~~

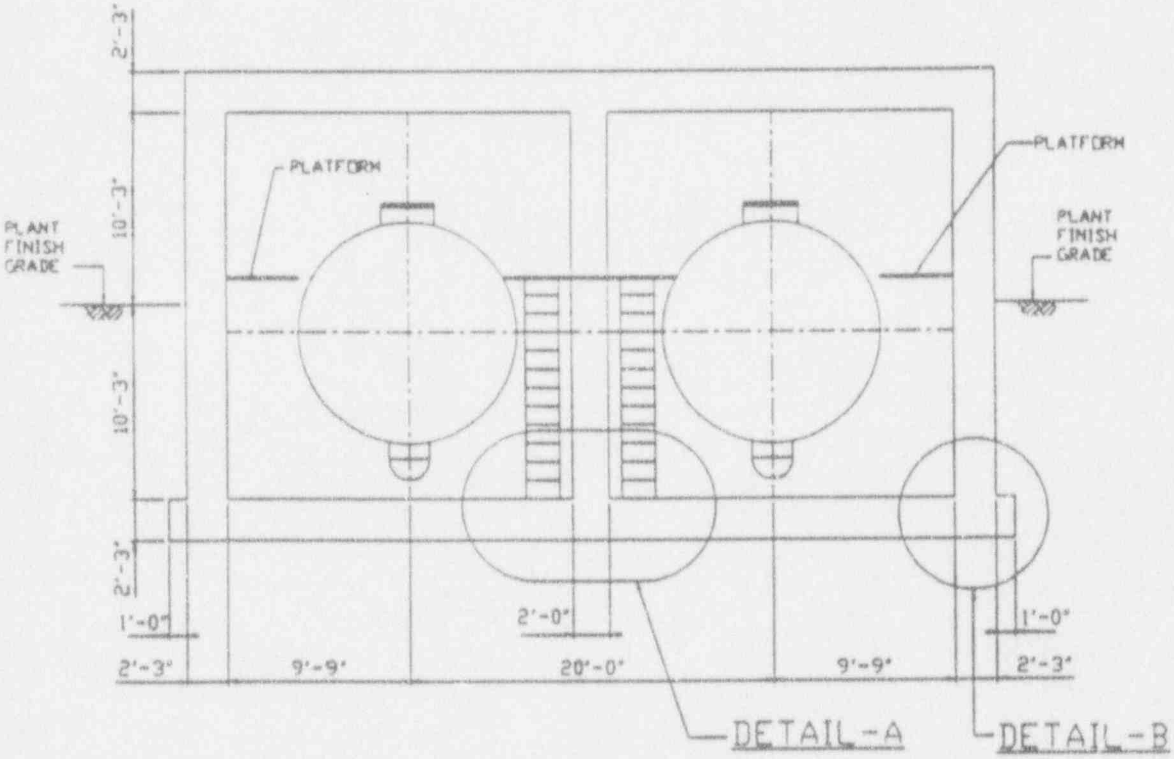
Roof:

The primary reinforcing for these two-foot thick roof slabs consist of a rectangular grid of #11 at 10 inches each way/each face, [i.e., 1.87 in^2/ft].

No transverse shear reinforcing is required.

7.3.5 CONCLUSION

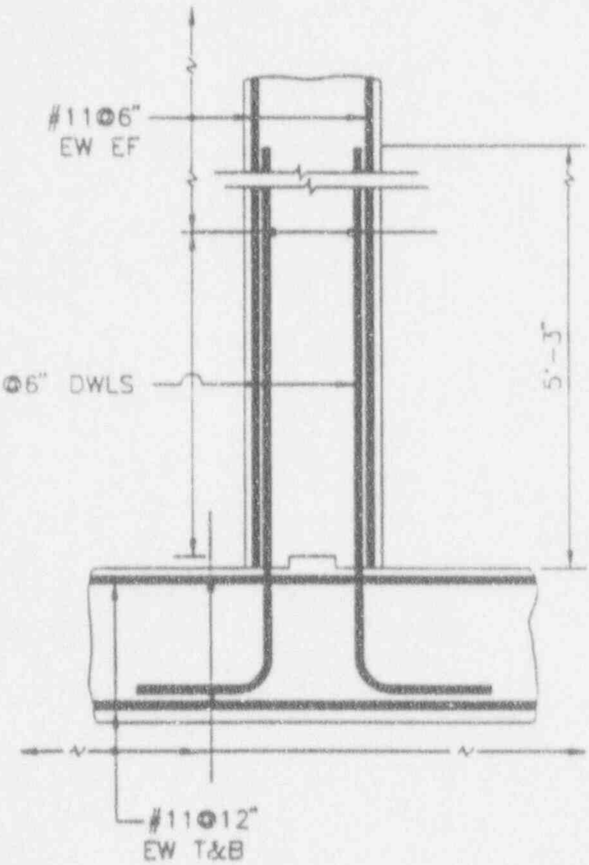
The concrete and reinforcing steel section strengths of the Component Cooling Water Tunnel are sufficient to resist the design basis load and load combination criteria specified in Sections 3.8A.11.7 and 3.8A.5.0. Typical reinforcing details are shown in Figures 3.8B-10 and 3.8B-11.



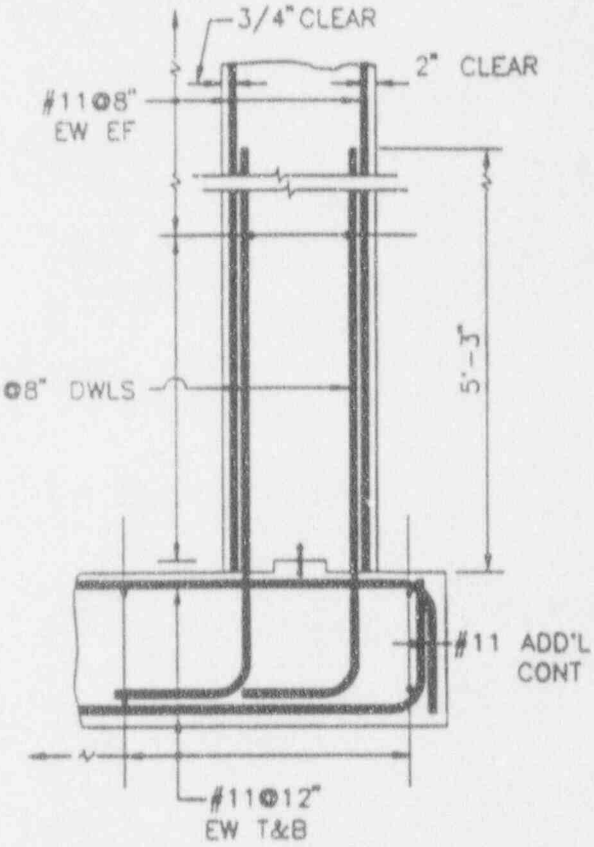
DIESEL FUEL STORAGE STRUCTURE

FIGURE 3.8B-5

~~SHEET 1 OF 2~~



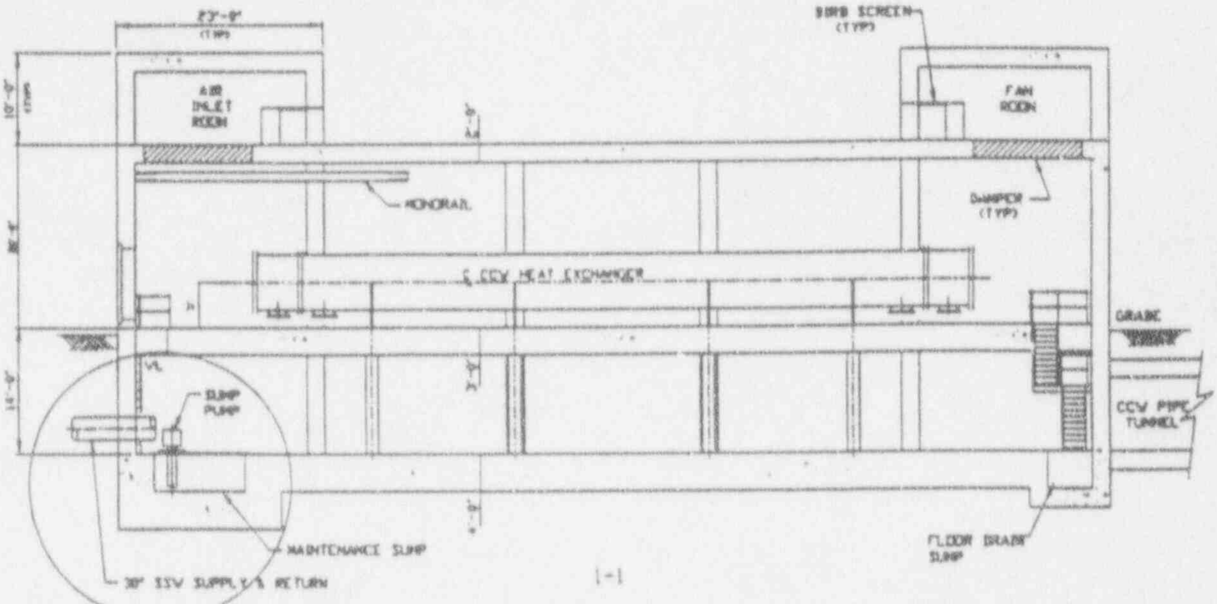
DETAIL-A



DETAIL-B

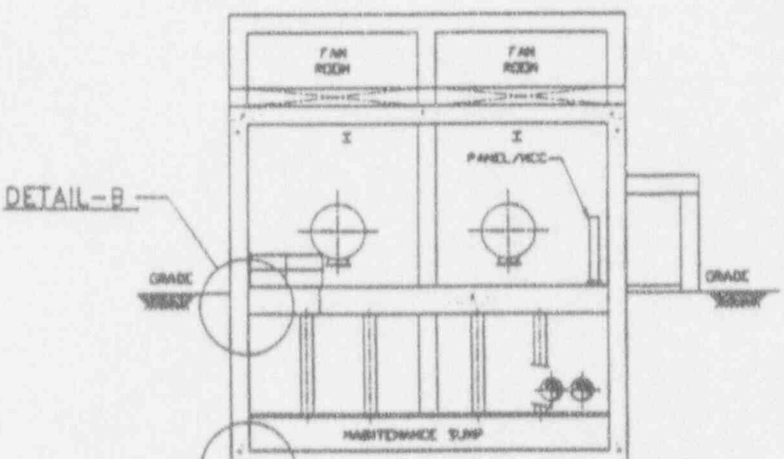
DEISEL FUEL STORAGE STRUCTURE
REINFORCING DETAILS

FIGURE 3.8B-6
~~SHEET 2 OF 2~~



SECTION LOOKING SOUTH

DETAIL-A



2-2

SECTION LOOKING EAST

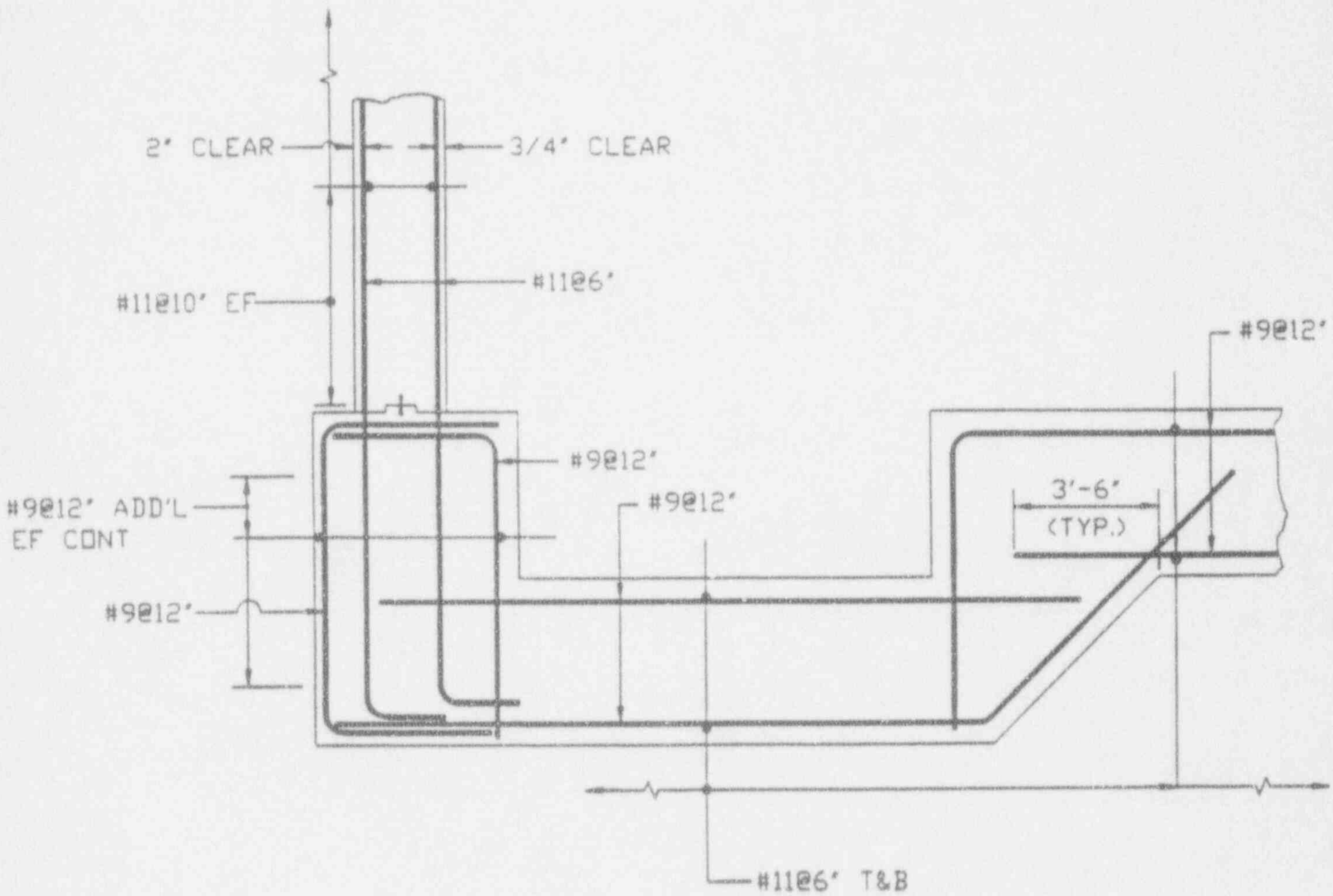
DETAIL-C

GENERAL ARRANGEMENT

CCW HEAT EXCHANGER STRUCTURE

FIGURE 3.8B-7

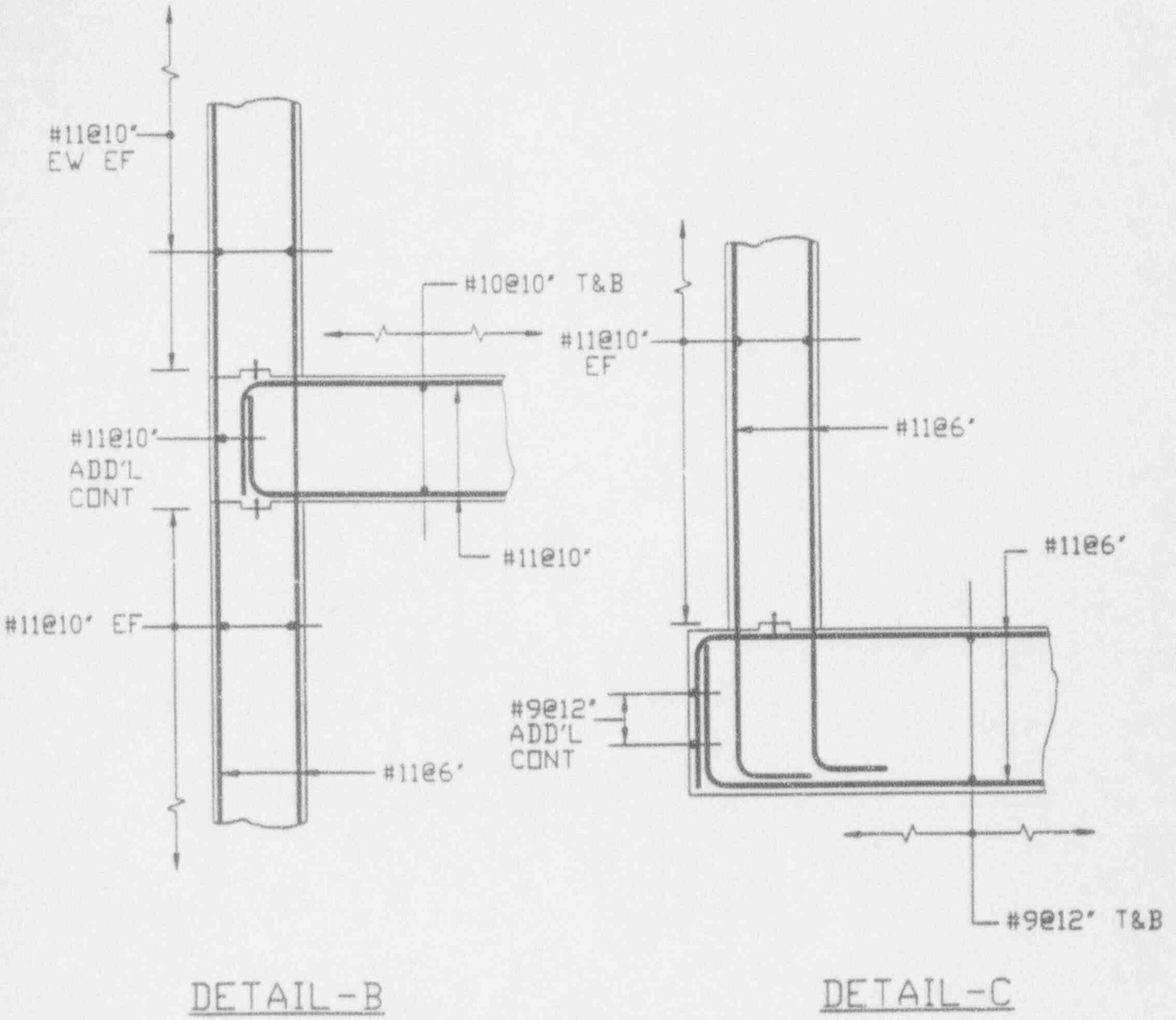
~~SHEET 1 OF 3~~



DETAIL-A

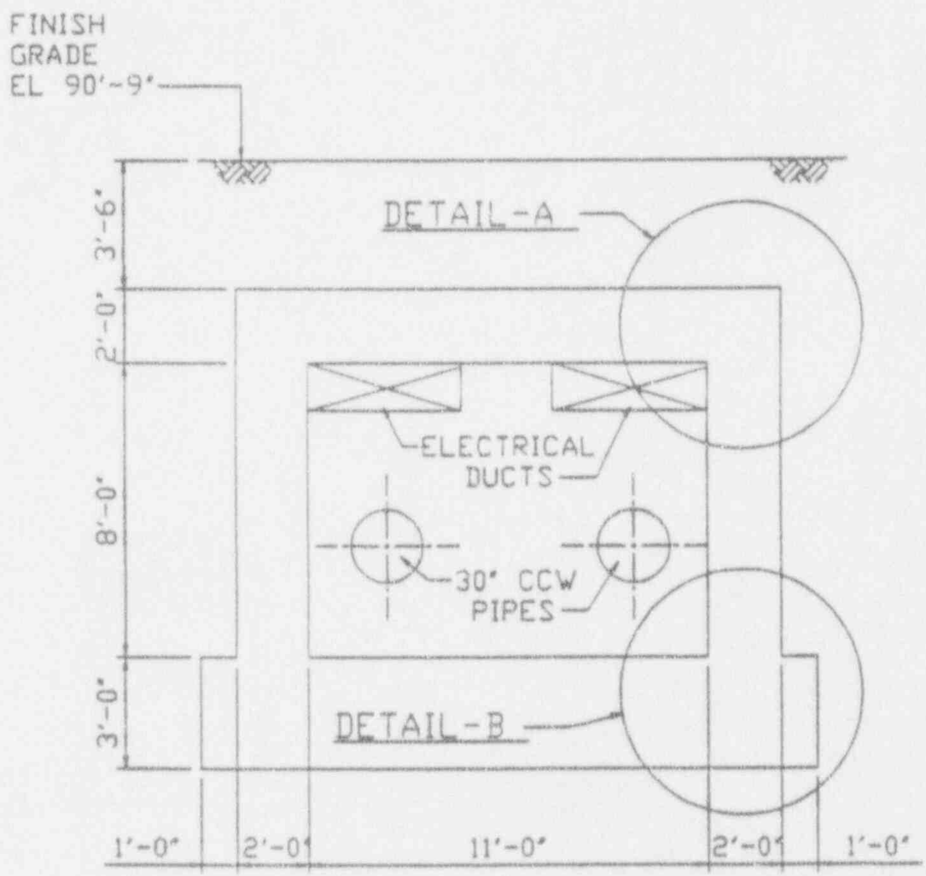
CCW HEAT EXCHANGER STRUCTURE
REINFORCING DETAILS

FIGURE 3.8B-8
~~SHEET 2 OF 3~~



CCW HEAT EXCHANGER STRUCTURE
REINFORCING DETAILS

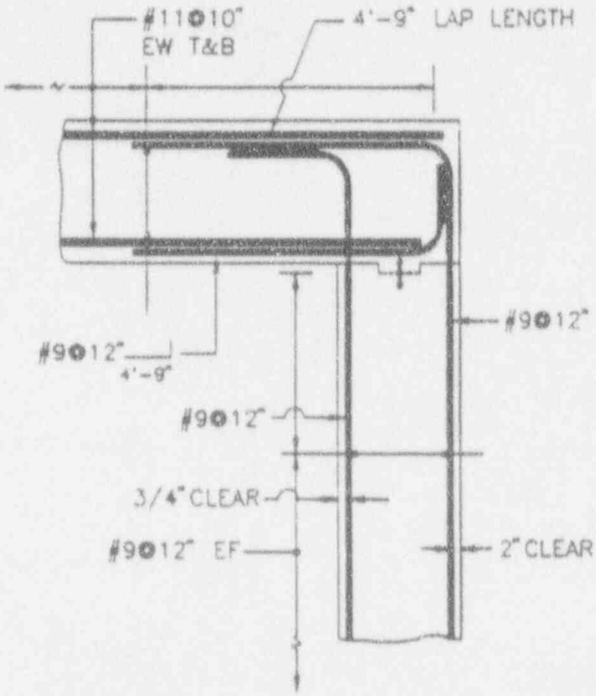
FIGURE 3.8B-9
~~SHEET 3 OF 3~~



COMPONENT COOLING WATER TUNNEL

FIGURE 3.8B-10

~~SHEET 1 OF 2~~



DETAIL-A

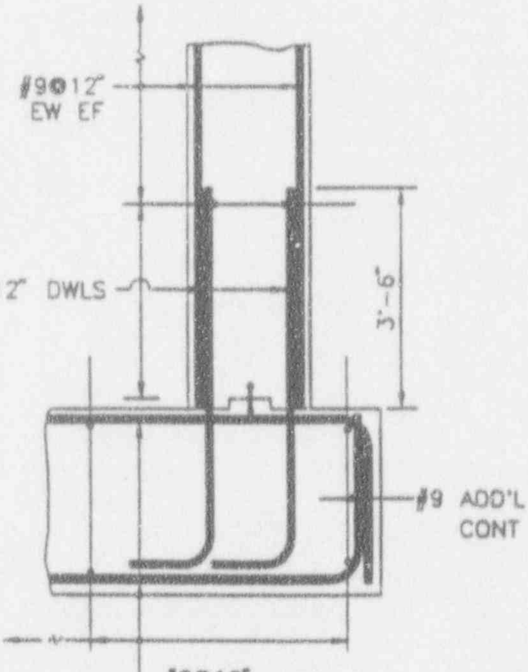


FIGURE 3.8B-11
~~SHEET 2 OF 3~~

DETAIL-B

COMPONENT COOLING WATER TUNNEL
REINFORCING DETAILS

ATTACHMENT 2

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On March 30, 1989, Combustion Engineering, Inc. tendered its application for certification of the System 80+ standard design with the U.S. Nuclear Regulatory Commission (hereinafter referred to as the NRC, the Commission, or the staff). The submittal was made in accordance with Appendix O, "Standardization of Design: Staff Review of Standard Designs," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Combustion Engineering, Inc., subsequently stated in a letter dated August 21, 1989, that its application may be considered as an application for design approval and subsequent design certification pursuant to 10 CFR 52.45. The application was docketed on May 1, 1991, and assigned Docket No. 52-002. Correspondence relating to 4
the application prior to this date were also addressed to Docket No. 50-370 and Project No. 675. This correspondence is listed in Appendix B to this report. In a letter dated May 26, 1992, Combustion Engineering, Inc. notified the NRC that they are a wholly owned subsidiary of Asea Brown Boveri, Inc., and the appropriate acronym for their company is ABB-CE. Therefore, the staff refers to Combustion Engineering, Inc. as ABB-CE throughout this report.

The NRC's licensing project managers assigned to the System 80+ standard design review are Mr. Thomas Wambach, Mr. Michael Franovich, and Mr. Stewart Magruder. They may be reached by calling (301) 492-7000 or by writing to the Office of Nuclear Reactor Regulation, Mail stop 11-H-3, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

ABB-CE's application, the Combustion Engineering Standard Safety Analysis Report Design Certification (CESSAR-DC), describing the design of the facility, was originally submitted on March 30, 1989. Subsequently, ABB-CE supplemented the information in CESSAR-DC through an amendment process. The most recent amendment, Amendment U, was submitted to the Commission on February 8, 1994. ABB-CE also submitted the System 80+ certified design

The staff is also performing a review of CESSAR-DC, final technical specifications (TS), CDM and this report to ensure that this information is internally consistent. Any inconsistencies or discrepancies will be resolved before issuance of the FSER. This is FSER Confirmatory Item 1.1-2.

Several references to ABB-CE reports are made in this report. Some of these reports contain information that has been authorized by the Commission to be exempt from public disclosure, as provided by 10 CFR 2.790. For each such report containing proprietary information, a nonproprietary version, similar in content except for the omission of the proprietary information, is provided to the NRC by ABB-CE and is also available at the NRC Public Document Room. Several references to ABB-CE reports throughout this advance FSER are made to the proprietary version only. The staff based its findings on the proprietary versions of these documents.

In its application, ABB-CE ~~committed to conform with~~ stated that the criteria included in the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor Program would be used as a guide for the design of System 80+. The Commission had requested that the staff evaluate any differences that the vendor designs have with the EPRI Utility Requirements Document (URD) in a staff requirements memorandum (SRM) dated December 15, 1989. On December 21, 1990, ABB-CE sent the staff a summary of the differences between its design and the EPRI URD. In the DSER, the staff identified an open item (DSER Open Item 1.1-1) for ABB-CE to address any System 80+ deviations from the EPRI URD. Subsequently, ABB-CE indicated that they were a principal participant in the development of the EPRI sponsored URD and continue to be involved with EPRI on changes to that document. Therefore, the design in CESSAR-DC remains consistent with the EPRI URD. The Commission designated this response to be acceptable in COMSECY-93-040, dated August 10, 1993. In a letter of January 7, 1994, ABB-CE stated that the System 80+ design was consistent with the EPRI URD. The staff finds this acceptable. On this basis, DSER Open Item 1.1-1 is resolved.

Plant-specific applicants who reference the System 80+ standard design in the future will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL), the staff will evaluate, for each plant-specific application that references

Steam Generator Design

1. An increased tube plugging margin
2. An increased boil dry time by increasing the secondary inventory
3. Corrosion resistant tubes
4. Higher steam quality
5. Increased manway size

Engineered Safety Systems Improvements

1. Increased SIS redundancy - four trains separated into quadrants
2. Increased ESF capacity relative to power
3. An in-containment refueling water storage tank, which eliminates automatic suction realignment and allows for reactor cavity flooding.
4. Safety depressurization system
5. Eliminated automatic injection by the residual heat removal (RHR) pumps
6. Eliminated chemical volume and control system (CVCS) safety functions
7. Higher pressure rating of RHR
8. Separate emergency and startup feedwater
9. Eliminate automatic isolation of EFWS

Containment Improvements

1. Larger margin to ultimate strength
2. Larger operating floor space
3. Larger free volume
4. Dual containment (pressure boundary plus a shield building)
5. Equipment hatch sized for SG replacement
6. Reactor cavity designed for severe accident mitigation

Instrumentation and Control System Improvements

1. Advanced control, alarm, and display systems
2. New control room design
3. Added APS

Steam Generator Design

1. An increased tube plugging margin
2. An increased boil dry time by increasing the secondary inventory
3. Corrosion resistant tubes
4. Higher steam quality
5. Increased manway size

Engineered Safety Systems Improvements

1. Increased SIS redundancy - four trains separated into quadrants
2. Increased ESF capacity relative to power
3. An in-containment refueling water storage tank, which eliminates automatic suction realignment
4. Safety depressurization system
5. Eliminated automatic injection by the residual heat removal (RHR) pumps
6. Eliminated chemical volume and control system (CVCS) safety functions
7. Higher pressure rating of RHR
8. Separate emergency and startup feedwater
9. Eliminate automatic isolation of EFWS

Containment Improvements

1. Larger margin to ultimate strength
2. Larger operating floor space
3. Larger free volume
4. Dual containment
5. Equipment hatch sized for SG replacement

Instrumentation and Control System Improvements

1. Advanced control, alarm, and display systems
2. New control room design based on modern technology
3. Added APS ?

APS is not an added system. System 80 had the same system with a different name. Design does differ.

by Section 52.47(a)(1)(iii) is provided in Chapter 2, and they are also discussed in Section 14.3. The staff's evaluation of the design-specific probabilistic risk assessment (Section 52.47(a)(1)(v)) is provided in Chapter 19. The evaluation of the inspections, tests, analyses, and acceptance criteria (ITAAC) required by Section 52.47(a)(1)(vi) is described in Section 14.3 of this report.

Interface requirements and representative conceptual designs (52.47(a)(1)(vii) through (ix)) are evaluated throughout Chapters 8 and 9 of this report, and are also discussed in Section 14.3. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, and the Commission's SRM on SECY-93-087, dated July 21, 1993, in its resolution of severe accident issues. The staff's evaluation of severe accident issues is provided in Section 19.2 of this report.

Section 52.47(a)(2) describes the level of design information needed to certify a standard design. The acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The SRM for SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," set forth the Commission's position on what level of design information was required for a certification application, and the staff has followed that guidance in preparing this document. ~~The staff determined that ABB-CE did not provide sufficient detail in CESSAR-DC for the following four areas of the review: pipe stress analysis, radiation shielding and airborne concentrations, instrumentation and controls (I&C), and control room design. Instead, the staff based its safety determinations for these areas of the design on the use of design acceptance criteria (DAC). The DAC are part of the CDM proposed for the System 80+ design~~. The staff's evaluation of the proposed CDM, including the DAC, is provided in Section 14.3 of this report.

Insert A

As part of its technical review, the staff made numerous RAI to provide sufficient bases for its safety findings, thereby meeting the requirement in Section 52.47(a)(3) to advise ABB-CE on the staff's requirements for additional technical information. An index of ABB-CE's responses to these RAIs is provided in Appendix B of this report. The staff's evaluation of the scope of

Insert A

To allow for technology improvemenbts and as-procured equipment characteristics, ABB-CE has proposed methods of analysis and corresponding acceptance criteria rather than design details. ABB-CE has proposed such methods and criteria for two design areas which span many systems and, therefore, has documented them in CESSAR-DC and in their Certified Design Material document as "process" ITAAC. The NRC staff prefers the terminology "Design Acceptance Criteria (DAC).

the design to be certified (Section 52.47(b)(1)) is provided in Section 1.2 of this report. The requirements set forth in Sections 52.47(b)(2) and (3) do not apply to the System 80+ standard design, because the System 80+ is an evolutionary reactor design~~, and also is not a modular design.~~ (which does not require prototype testing) and because System 80+ is designed as a single unit (no safety systems would be shared at a multi-plant site). The staff used the safety standards set forth in Section 52.48 for its technical review of the System 80+ standard design. In addition to these safety standards, the staff also followed Commission guidance provided in the SRMs for all applicable Commission papers, including those identified in Table 1.5. As a result of this guidance, the staff proposed design-specific regulations and justified exemptions from existing regulations in order to complete the framework of safety standards. An index to these safety standards is provided in Section 1.8 of this report.

Subsequent to the completion of the staff's review of the CESSAR-DC and CDM, ABB-CE will prepare a design control document (DCD). The DCD will consist of the CDM and Tier 2 information as described in Section 14.3 of this report. Applications that reference the certified System 80+ design will be required to conform with the DCD. The DCD will be available for public inspection at the NRC's Public Document Room when the proposed rule for design certification is published in the Federal Register.

1.6 Summary of Open Items

The staff has identified no open items in its review of the System 80+ design in this report. In the DSER, the staff identified a number of unresolved or open issues as a result of its review up to the DSER cutoff date of May 8, 1992. All of the DSER open items, and all of the unresolved items identified in the staff's review subsequent to May 8, 1992, up to and including Amendment U, have been resolved as described throughout this report.

1.7 Summary of Confirmatory Items

In the DSER, the staff identified a number of confirmatory items as a result of its review up to the DSER cutoff date of May 8, 1992. All of the DSER confirmatory items were resolved as described throughout this report.

17.3	Applicable regulation for reliability assurance program.
19.1.2.2	Applicable regulation for core debris cooling.
Section	Description of Applicable Regulation
19.1.3.2.1	Applicable regulation for containment performance.
19.1.4.1	Applicable regulation for seismic margins.
19.2	Applicable regulation for high-pressure core melt ejection.
19.2	Applicable regulation for equipment survivability.
19.3	Applicable regulation for shutdown risk.

1.9 Interface Requirements

Section 52.47(1)(1)(vii) requires interface requirements that must be met by the site-specific elements of the non-certified portion of the plant design, such as the ultimate heat sink. The scope of the System 80+ design is discussed in CESSAR-DC Section 1.2 and is illustrated in CESSAR-DC Table 1.2-1. A discussion of the certified and non-certified portions of the System 80+ design is provided in Section 1.2 of this report. CESSAR-DC Table 1.9 provides an index of the interface requirements for the System 80+ design.

1.10 Combined License Action Items

COL applicants and licensees who reference the certified System 80+ standard design in the future will be required to satisfy the requirements and commitments in the DCD. Also, certain requirements and commitments are identified in the CESSAR-DC as "COL License Information," and in this report as "COL Action Items." These COL action items relate to programs, procedures, and issues that are outside of the scope of the certified design review. An applicant for a COL will be required to address each of these items in its application.

ABB-CE included the list of COL action items in Chapter 1 of the CESSAR-DC and provided an explanation of the items in the applicable sections of the CESSAR-DC. In the DSER, the staff had identified a number of COL action items from its review. ABB-CE incorporated the COL action items that had been identified by the staff in the DSER and referred to these items in Table 1.10-1 of the CESSAR-DC as "COL license information." The following issues have been identified by the staff as COL action items that should be added to CESSAR-DC Table 1.10-1, and updated in the appropriate sections of CESSAR-DC. This is FSER Confirmatory Item 1.10-1.

We are working on consistency of this list, the whole FSER and CESSAR.

Item Number	Description
2.3.1-1	Regional climatology.
2.3.3-1	Onsite metrological measurements program.
6.2.4-1	Containment isolation valve location and pipe size defined in accordance with General Design Criteria 64.
9.2.5-1	Protected area perimeter should not abut or cross ultimate heat sink body of water.
9.5.1.2.1.2-1	Procedures and training for using transfer switches.
11.2.1-1	Provide setpoints for radiation monitors in plant-specific offsite dose calculation manual.
11.5.1-1	Provide operation and maintenance manual for monitoring and sampling liquid and gaseous process and effluent streams.
13.2-1	TMI I.A.4.2, "Long-Term Training Simulator Upgrade."

See next page - FAX

300-580 L/min (75-145 gpm)

MSL leakage detection along with the commitment in the TSs is acceptable. The leakage rate of 38 L/min (10 gpm) is based on ABB-CE's calculation of the maximum leakage rate from a crack opening under NOP loadings. ABB-CE calculated this maximum leakage rate to be approximately 700-800 L/min (175-200) gpm. There is sufficient margin by comparing the calculated 700-800 L/min (175-200 gpm) with the TS limit of 40 L/min (10 gpm). Therefore the staff concludes that the MSL leakage detection for application of LBB to the MSLs in the System 80+ design is acceptable.

(5) The acceptance criteria in the PEDs were developed in part by applying a margin of √2 on loads utilized in the leakage-size flaw stability analyses. This margin of √2 on loads is consistent with NUREG-1061, Volume 3 criteria for the method of combining the load components and hence acceptable.

(6) Similar to (5), the acceptance criteria in the PEDs were also developed on the basis of stability analyses of flaws 2 times the leakage-size flaws for the loads considered. This margin of 2 on leakage-size flaw is consistent with NUREG-1061, Volume 3 criteria and hence acceptable.

Relative to the development of the PEDs in (5) and (6) preceding, the staff would note that the boundaries of the regions of allowable loading in these diagrams were developed on the assumption that the boundaries were piecewise linear. Additional calculations performed by ABB-CE verified that this assumption was essentially valid and also conservative. Moreover, torsional moments were not considered in the flaw stability evaluations since their effects were small relative to those due to bending moments. Furthermore, two evaluation diagrams are needed for the SL since thermal stratification effects are recognized as significant for the SL. Similar effects in the DVI line were small relative to the other loads and were neglected.

COMBUSTION ENGINEERING

Date 3/23/94

Time AM

TELEPHONE CALL RECORD

From D. A. Peck of CE

To David Tarao of NRC

Contract Name and Number CESSAR-DC

Route: L D Gerdes

J J LaRussa

S E Ritterbush

Subject: FSER Comments

I advised NRC that the values shown on page 3-76 of the FSER were incorrect. The first # numbers "700-800 L/min (175-200) gpm" should be "300-580 L/min (75-145) gpm" two places. The original values were mentioned off-the-top-of-the-head in a phone call, and later revised in a second phone call, after some calculations were performed.

Also, I advised that the LBB-PED Figures 3.9A-30 to 3.9A-36 will be revised in the next Rev of CESSAR to all be consistant with the use of PICEP, as stated in the text.

David did not seem concerned that there would be any problems

C-E 0010042 (Rev. 6/85)

Written By D Peck

operating limits. This system, which consists of software executed on the plant computer, utilizes the output of the incore detector system to synthesize the core average axial power distribution. Rod positions taken from the control rod position indication system, together with precalculated radial peaking factors, are used to construct axially dependent, radial power distributions. By using this information, together with measured primary coolant flow, pressure, and temperature, the COLSS establishes the margin to the operating limits on maximum linear heat generation rate and minimum DNBR. The system also monitors azimuthal flux tilt and total power level and generates an alarm if any of these limits are exceeded. The margins to all of these limits except azimuthal tilt are continuously displayed to the operators; the tilt can be displayed at the request of the operator. The operator monitors these margins and takes corrective action if the limits are approached. These actions include improving the power distribution by moving full-strength or part-strength rods, reducing power, or changing thermal-hydraulic conditions, that is, coolant inlet temperature and primary system pressure.

CEN-356(V)-P-A, Rev. 01-P-A "Modified Statistical Combination of Uncertainties".

A description of the COLSS algorithms and an uncertainty analysis of the calculations performed by the COLSS is presented in CE Topical Report ~~CENPD-169-P, "COLSS Assessment of the Accuracy of Pressurized-water Reactor (PWR) Operating Limits as Determined by the Core Operating Limit Supervisory Systems.~~" The staff reviewed this report and found the methods employed in COLSS to determine power distributions are acceptable. The COLSS is currently used at ANO (Unit 2), San Onofre (Units 2 and 3), Waterford (Unit 3), and Palo Verde (Units 1, 2, and 3).

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. ABB-CE presents calculated values of the coefficients in the CESSAR-DC and has also evaluated the accuracy of these calculations. The staff reviewed the calculated values of reactivity coefficients and concludes that they adequately represent the full range of expected values.

As an alternative, ABB-CE could discuss the applicability of the previously submitted and approved documents for the System 80 CPC/CEAC design to the System 80+ design. This was identified as DSER Open Item 4.4.4-1.

In response, ABB-CE revised CESSAR-DC 7.2.1.1.2.5 (Amendment R) to indicate that the software design of the CPC/CEAC system is described in References ~~10~~ 1 and 2 and 4 through 8 ~~through 16~~, and has been reviewed and approved by the NRC in References ~~15~~ 7 through ~~21~~ 13. The COL application and ABB-CE will follow the procedures described in References 22 and 23 for all changes to the algorithms, data base constants and data block constants for the CPCs and CEACs. The staff finds that the procedures documented in References 22 and 23 were previously approved by the NRC. The overall CPC/CEAC software implementation, which is to translate the system functional requirements into modules of machine executable code and to integrate these modules into a real time software system, is verified through the Phase I and Phase II software verification test. The scope of testing will include generation of plant-specific data base document, generation of appropriate test cases and acceptable criteria, and test reports. Phase I testing is to be performed on the DNBR/LPD calculation systems to verify that CPC/CEAC system software modifications have been properly implement. Phase II testing is performed on the CPC/CEAC system to verify that CPC and CEAC software modifications have been properly integrated with the CPC and CEAC software and system hardware, and to provide confirmation that the static and dynamic operation and the integrated system as modified is consistent with that predicted by design analyses.

Testing of the CPC/CEAC software for each license applicant referencing the System 80+ design certification will be considered complete with the formal issuance of (1) CPC/CEAC data base document, (2) the Phase I test report, and (3) the Phase II test report. These documents are plant specific and will be reviewed individually for each license application referencing the System 80+ design certification. DSER Open Item 4.4.4-1 is, therefore, reclassified to COL Action Item 4.4.4-1.

4/4/94

thermal-hydraulic analyses using analytical methods, DNBR correlations, and the safety limit DNBR that the staff previously approved. Therefore, the staff concludes that the thermal-hydraulic design of the System 80+ design core provides appropriate thermal margin to assure that SAFDLs are not exceeded during any conditions of normal operation and AOOs, and thus, conforms to the requirements of GDC 10 and is acceptable. The staff's evaluation of the calculated DNBRs during AOOs is included in Sections 15.1 through 15.3 of this report.

Each COL applicant referencing the System 80+ design certification has overall responsibility for the startup test program. However, the CESSAR-DC defines ABB-CE's participation and provides guidelines to the reference plants for pre-operational and initial startup test program in accordance with RG 1.68 to measure and confirm thermal-hydraulic design aspects. The evaluation of startup testing program is included in Section 14 of this report.

References for Section 4.4

1. "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-A, April 1986.

2. "TORC Code - Verification and Simplified Modeling Methods," CENPD-206-P-A, June 1981.

3. "CETOP-D Code Structure and Modelling Methods for Arkansas Nuclear One Unit 2," CEN-214-A-P, July 1982.

4. Modified "Statistical Combination of Uncertainties," ~~CEN-139-A-P, November 1980~~ CEN-356(V)-P-A, Rev. 01-P-A, May 1988.

5. "Critical Heat Flux Correlations for CE Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-A-P, September 1976.

6. "Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Power Distribution," CEN-207-A-P, June 1976.

The overpressure protection design for the System 80+ plant at power operating conditions complies with the guidelines of SRP Section 5.2.2 and the requirements of GDC 15 and, therefore, is acceptable for design certification.

5.2.2.2 Overpressure Protection During Low Temperature Operation

Guidelines in SRP Section 5.2.2 state that the system for overpressure protection during low temperature phases of plant operation should be designed in accordance with the requirements of Branch Technical Position (BTP) RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures."

The low temperature overpressure protection (LTOP) system for the System 80+ design is provided by the spring-loaded liquid relief valves in the SCS. One SCS liquid relief valve is provided in each of the two SCS pump suction lines. These two valves are set at a pressure low enough to prevent violation of the 10 CFR Part 50, Appendix G heatup and cooldown curves should a pressure transient occur during low temperature operations. For LTOP considerations, two types of events are considered as the design basis events. These events are: (1) the mass addition transient caused by charging and safety injection flows following an inadvertent safety injection actuation, and (2) the heat addition transient caused by the restart of a reactor coolant pump. ABB-CE determined the pressure set point and flow capacity for the SCS relief valves based on the mass addition transient, which has been demonstrated as the limiting case, and which results in the highest pressure increase. The mass addition transient analysis was performed assuming simultaneous operation of four safety injection pumps and one charging pump with the letdown system isolated. All pressurizer heaters were assumed to be operating from a water solid condition to maximize the pressure increase. The result shows that the peak pressure is ~~4190~~ 4275 kPa (~~608~~ 620 psia) for a relief valve with the pressure set point of 3760 kPa (545 psia) and flow capacity of 19000 L/min (5000 gpm) of water. ABB-CE specified 19000 L/min (5000 gpm) for each of two valves as a rated relief capacity. This rated relief capacity will meet the required relief flow for the worst transient.

CHANGED IN AMENDMENT U

LTOP enable temperatures for initiation of the LTOP system can be determined by following BTP RSB 5-2 guidance for LTOP. According to the BTP 5-2 guidance, the enable temperature is the water temperature corresponding to metal temperature of at least reference nil-ductility transition temperature (RT_{NDT})+ 50 °C (90 °F) at vessel beltline location (either 1/4t or 3/4t) that is controlling in the Appendix G (to section III of the ASME Code) limit calculations. Conforming with BTP 5-2 guidance, ABB-CE determined that the disable temperature is 101 °C (214 °F) for heatup with the heatup rate of 22 °C/hr (40 °F/hr) or less and the enable temperature is 86 °C (187 °F) for the controlling (isothermal) cooldown of 0°C/hr (0°F/hr). ~~with the cooldown rates less than 56 °C/hr (1[illegible] °F/hr)~~. However, ABB-CE proposed LTOP disable and enable temperatures defined by intersections of the 10 CFR Part 50, Appendix G heatup and cooldown curves and the pressurizer safety valve set point of 17200 kPa (2500 psia). Consistent with Technical Specifications 3.4.3 and 3.4.11, the heatup rate must be less than 22 °C/hr (40 °F/hr). ~~and the cooldown rate must be less than 56 °C/hr (100 °F/hr).~~ The disable and enable temperatures are thus determined to be 143 °C (290 °F) for heatup and 126 °C (259 °F) for cooldown, respectively. The proposed disable and enable temperatures exceed the required temperatures in accordance with BTP RSB 5-2 for LTOP with sufficient margin in overpressure protection, and, therefore, are acceptable.

For temperatures above the LTOP enable temperature, overpressure protection is provided by the pressurizer safety valves. Before entering the low temperature region for which LTOP is necessary, administrative controls require operators to decrease the RCS pressure below the maximum pressure allowable for SCS operation. The SCS will be aligned whenever the RCS is at low temperatures and the reactor vessel head is secured, or until an adequate vent has been established.

System design criteria required by the staff include: the mitigating system must meet single-active failure criteria; the system must be capable of being tested; and the system must be capable of functioning following loss-of-offsite power. ABB-CE has met all the design criteria for LTOP. This provides assurance that the temperature-pressure limit presented in Appendix G of 10 CFR Part 50 will not be exceeded during any transients.

DSER Open Item 5.2.2.2-1, regarding the use of cobalt containing alloys, such as Stellite, is resolved as follows:

Stellite is a cobalt-based alloy. Activation of cobalt is a concern relating to the radioactivity in current nuclear plants. Therefore, ABB-CE should avoid the use of cobalt for ALARA considerations. In CESSAR-DC Section 5.2.3.2.2, "Materials of Construction Compatibility With Reactor Coolant," ABB-CE states that cobalt-based alloys will be avoided except in cases where no proven alternative exists. Cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt base) type alloys, although under development, have not been fully demonstrated to have the usability of the Stellites at this time. The NRC staff encourages the COL applicant to monitor the continuing development of cobalt-free hardfacing alloys in nuclear power plant systems to reduce future radiation exposure of personnel. On this basis, Open Item ~~5.2.2-1~~ is resolved.
5.2.2.2-1

The COL applicant should determine the LTOP enable temperature based on plant-specific material properties and pressure-temperature limit curves. This is COL Action Item 5.2.2.2-1.

5.2.2.3 Pressurized Thermal Shock

Pressurized thermal shock (PTS) events are system transients in a pressurized water reactor that can cause severe overcooling followed by immediate repressurization to a high level. The thermal stresses, caused when the inside surface of the reactor vessel cools rapidly, combine with the pressure stresses to increase the potential for fracture if an initiating flaw is present in low toughness material. This material may exist in the reactor vessel beltline, adjacent to the core, where neutron radiation gradually embrittles the material over time. The chemical composition of the steel is an important determining factor regarding the degree of embrittlement. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," (56 FR 22300; May 15, 1991) establishes a PTS screening criterion (RT_{PTS}) below which no additional action is required for protection from PTS events.

5.4.3 Shutdown Cooling System

ABB-CE described the shutdown cooling system (SCS) in CESSAR-DC Section 5.4.7. The SCS is designed to remove heat from the reactor coolant system (RCS) during a reactor shutdown after the RCS temperature and pressure have been reduced to approximately 177 °C (350 °F) and 3150 kPa (450 psia). The SCS is capable of reducing the RCS temperature to the refueling conditions and maintaining this temperature until the plant is started again.

The SCS also performs the following functions:

- transfers RCS fluid to the chemical and volume control system (CVCS) for purification of RCS fluid during SCS operation

- transfers refueling pool water back to the inside-containment refueling water storage tank (IRWST) following refueling operations

INSERT

- provides a backup to the containment spray system for IRWST heat removal during accident conditions

- provides RCS low temperature overpressure protection

Relate to CESSAR Amendment I which was clarified by "insert."

The staff reviewed the SCS for the System 80+ design in accordance with SRP Section 5.4.7. The design acceptance criteria are stated in BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System." ABB-CE's approach to meeting these requirements is discussed below.

5.4.3.1 Functional Requirements

In SRP Section 5.4.7, the staff requires the SCS design to meet GDC 1 through 5. Compliance with GDC 1 through 4 is addressed in Chapter 3 of this report. ABB-CE has satisfied GDC 5, since components of the SCS will not be shared between units.

When the SCS is in operation, the system takes its suction from each hot leg via a system of parallel lines and valves forming redundant trains. From the

INSERT

- provides for cooling of the IRWST during post-accident feed and bleed operations utilizing the safety injection system and the safety depressurization system.

discharge of the two SCS pumps, a portion of the reactor coolant is circulated through two SCS heat exchangers, which are cooled by component cooling water. The reactor coolant then returns to the RCS through safety injection system (SIS) direct vessel injection (DVI) nozzles. During normal shutdown, when non-safety-related equipment and offsite power are available, decay heat is removed from the core by the main feedwater system, the steam bypass system, and reactor coolant pump (RCP) circulation system. During emergency shutdown, when non-safety-related equipment and offsite power are not available, decay heat is removed by RCS natural circulation with the steam generators (SGs) as the heat sink. The steam produced in the SG shell side can be removed through the safety-related SG safety valves and atmospheric steam dump valves to vent vaporized secondary coolant. Secondary coolant makeup is the emergency feedwater pumped from two safety-grade emergency feedwater storage tanks.

Do you want to add main spray when the RCPs are ~~available~~ on?

RCS depressurization can be achieved by one of three systems: (1) auxiliary pressurizer spray, (2) rapid depressurization system (RDS), and (3) reactor coolant gas vent system (RCGVS), and (4) main spray when the RCPs are on. The auxiliary pressurizer spray is not designed to satisfy single-failure criteria and is not credited in the BTP RSB 5-1 analysis. Both RDS and RCGVS are safety-grade systems. The RCGVS is designed for RCS depressurization for design basis events, and the RDS is used to mitigate consequences of a beyond design basis event. The staff evaluated the design of the RCGVS and RDS in Section 6.7 of this report.

~~ABB-CE asserted that the~~ P - The RCGVS is sufficient to supplement or complement ~~substitute for~~ the auxiliary pressurizer spray to accomplish depressurization during natural circulation conditions.

Two separate trains for each unit provide redundancy in the SCS. Each train is powered by an emergency diesel generator. No single active failure to the SCS can prevent at least one complete train of the SCS from being brought on line from the control room during normal plant cooldown, a transient, or an accident.

Before SCS initiation, the safety injection tanks must be secured or vented (from the control room) to prevent overpressurization of the SCS. During the cooldown, adding borated water (boration) to the RCS controls core reactivity.

For cold shutdown, boron is added to the RCS using the SIS. The source of borated water is the fluid in the IRWST. As opposed to the System 80 design, the System 80+ design does not rely on the CVCS as a means for boron injection to meet BTP RSB 5-1 requirements.

Item G of BTP RSB 5-1 requires that a seismic Category I emergency feedwater supply be provided with sufficient inventory to permit operation at hot shutdown conditions for at least 4 hours, followed by a cooldown to the conditions permitting operation of the SCS. The emergency feedwater needed for the cooldown is based on the longest cooldown time needed with either onsite or offsite power available and with the worst ~~with an assumed~~ single failure. ABB-CE included two safety-grade emergency feedwater trains in the System 80+ design. Each train includes one turbine-driven pump and motor-driven pump. The emergency feedwater system contains two safety-grade emergency feedwater storage tanks. Each tank has a minimum condensate volume of 1.3×10^6 L (350,000 gal) available for safe plant cooldown.

During the course of the review, the staff asked ABB-CE to submit the results of an analysis to demonstrate that the System 80+ plant can achieve cold shutdown during natural circulation conditions per the assumptions specified in BTP RSB 5-1. In the response to RAI Q440.51, ABB-CE referenced the CE System 80 report of August 12, 1983 (letter LD-83-074; Docket No.: STN 50-470F), as applicable to the System 80+ design. The CE report of August 12, 1983, contains an analysis of full natural circulation cooldown (NCC) from hot standby conditions to temperatures and pressures that permit the initiation of the SCS. The analysis was performed using only safety-related equipment concurrent with a loss-of-offsite power and the worst assumed single ~~active~~ failure (i.e. failure of one diesel generator to start). The results of this analysis indicate that the total time required to take the System 80 plants from hot standby conditions to the SCS initiation condition is approximately 10.5 hours. This time includes maintaining the plant in hot standby for 4 hours before commencing a cooldown.

In the report of August 12, 1983, for the System 80 analysis, the design used the safety-related auxiliary spray and charging systems for RCS depressurization and RCS inventory control. The safety-related RCGVS was not credited in the analysis. During this simulated cooldown process, approximately

832,800 L (220,000 gal) of condensate water are required, which is less than the minimum 1.1×10^6 L (300,000 gal) design condensate water capacity. ABB-CE compared the operational status of systems and equipment assumed in the analysis and indicated that the availability of the system and components for System 80+ is equivalent to System 80, and the capacities of those components for plant cooldown are equivalent to or better than that for System 80. In the System 80+ design, the auxiliary spray and charging systems are not ~~designed to be~~ single failure proof and therefore ~~are~~ not creditable for the BTP RSB 5-1 analysis. However, under the assumption of loss-of-offsite power, two trains of the high pressure safety injection system (HPSI) are available for System 80+ as opposed to one train of the HPSI system for System 80. ~~ABB-CE claimed that the~~ The flow capacity of the additional HPSI train for System 80+ exceeds that of the charging pumps credited in the System 80 analysis for the RCS coolant makeup. In the System 80 natural circulation cooldown analysis, the auxiliary spray was used to rapidly depressurize the ~~pressurizer~~ RCS. ~~once the steam bubble was removed for System 80.~~ ~~This~~ The RCS depressurization ~~can be~~ was conducted for System 80+ via the RCGVS, which has greater capability for depressurization than the System 80 auxiliary spray system. In addition, the condensate storage capability has more than doubled from the System 80 to System 80+ design. Thus, ~~ABB-CE asserted that~~ the August 12, 1983, analysis for System 80 bounds the System 80+ design.

¶ ~~ABB-CE's justification was reasonable qualitatively. However, the~~ The staff was concerned about the thermal-hydraulic response during depressurization by using the RCGVS. Also, since the makeup water is demanded at relatively high RCS pressures during the plant cooldown, the safety injection pumps may not be as effective as the charging pumps for the RCS inventory ~~makeup~~ control. The staff determined that ABB-CE had not adequately demonstrated the applicability of the August 12, 1983, analysis to the System 80+ design. Therefore, ABB-CE ~~was required~~ agreed to submit the results of analysis demonstrating that System 80+ is capable of achieving cold shutdown in accordance with the assumptions specified in BTP RSB 5-1.

In response to the staff's request, ABB-CE performed a NCC analysis by using only safety-grade equipment with an assumption of the concurrent loss of offsite power and the worst ~~a~~ single failure. ~~event~~. The results are presented in Amendment N to Appendix 5D of the CESSAR-DC. The previously NRC approved LTC code was used by ABB-CE to perform the analysis. The NCC sequence relied on the guidance in ABB-CE's System 80+ emergency operating ~~procedures~~ guidelines ~~(EPG, CEN-152)~~ as follows:

1. Following the reactor trip, the operator manually controls the atmospheric dump valves (ADV) and emergency feedwater flowrate ~~pumps~~ to restore the plant at hot standby conditions.

2. The operator throttles two of the four safety injection system pumps (only two trains are available because of the assumed single failure of one diesel generator) for RCS boration and inventory control.

3. After the four hour hot standby period, the operator uses the ADVs to initiate a cooldown with a rate less than that allowed ~~limited~~ by the TSs.

4. The operator uses the safety-grade equipment for RCS pressure and inventory control: the pressurizer vent system for depressurization; the reactor vessel ~~RCS~~ vent system for passive steam removal from the reactor upper head; and SI pump throttling for boron and RCS inventory control.

5. The operator controls the cooldown to maintain the pressurizer level and RCS subcooling within the ranges consistent with the ~~EPG~~ EOG guidance.

The analytical results indicate that the SCS entry conditions can be achieved within 10 12 hours (including four hours at hot standby conditions). The staff has reviewed ABB-CE's NCC analysis. The staff finds that the previously approved LTC code was used for analysis and only the safety-grade equipment was credited for natural circulation cooldown. The analytical results show that the SCS entry conditions have been achieved with the total emergency feedwater usage of less than 35 percent of the minimum available capacity. The staff concludes that the analysis is acceptable. BTP 5-1 requires ABB-CE to demonstrate the NCC capability by analysis and test. The staff believes that the use of RCGVS and SIS may result in a more complicated evolution than that demonstrated by use of auxiliary spray and CVCS. The staff's conclusion is based upon the SI pump curve which shows a rapidly increasing flow rate as pressure decreases, and the use of two vent valves for pressure control as

Figure 6.3.2.1C correctly reflects the valve position and indications for each isolation valve. On this basis, Open Item 5.4.3.2-1 is resolved.

The design of interlocks will not automatically close the isolation valves in the event of an RCS pressurization during shutdown cooling. This design of interlocks is to prevent a loss of decay heat removal capability due to inadvertent closure of the isolation valves during SCS operation. To respond to DSER Open Item 5.4.3.2-2, ABB-CE indicated that the System 80+ design has the following features for prevention of SCS overpressurization:

1. An alarm will be provided for each of the SCS suction isolation valves located inside containment. This alarm is to warn control room operators when rising RCS pressure approaches the SCS operating pressure limit and isolation valves are not closed.

2. Valve position indication will be provided in the control room for all SCS suction isolation valves located inside containment. The power for the indicators is supplied from a separate source such that the position indication is not effected by power interruption to the operator.

3. The response guidelines as described in CESSAR-DC Section 5.4.7.2.6 ~~5.4.7.2.6.1~~ will be provided to direct operator actions during shutdown cooling for overpressurization protection of the SCS.

NO SUCH SECTION
5.4.7.2.6 = "MANUAL ACTIONS" would seem to be appropriate

4. The design pressure of the SCS will be increased to 6205 kPa (900 psi). This design pressure meets the required ultimate rupture strength equal to full RCS pressure as specified in SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," to reduce risk of intersystem LOCA events (see Section 20, of this report Issue 105, for the evaluation of intersystems LOCA events).

Based on the staff's review, the staff finds that the design features discussed above provide reasonable assurance of overpressure protection for the

Item E of BTP RSB 5-1 states that "the isolation valve operability and interlock circuits must be designed as to permit on line test when operating in the RHR modes." ABB-CE has retracted this deviation and committed to the conformance of the System 80+ design to these testing requirements. Testing of the SCS suction isolation valve circuitry is provided in CESSAR-DC Sections 7.6.6.2 and 14.2.12.1.21.
7.6.2.2

The SCS design meets the isolation requirements of BTP RSB 5-1 and is acceptable. On this basis, Open Item 5.4.3.2-3 is resolved.

Intersystems LOCA

{Chip Coleman Reviewed: Para 2 The "must" and "should" implies "openness" Para 3 implies states resolution}

SECY-90-016 specifies the staff's position on protection against the possibility of a loss-of-coolant accident (LOCA) occurring outside the containment for those systems linked to the RCS. The staff position is that future advanced light water reactor (ALWR) designs should reduce the possibility of a LOCA outside the 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. The "extent practicable" phrase is a realization that all systems must eventually interface with atmospheric pressure, and that, for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to the URS 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 packing, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drains and vent lines. The designer 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 practical to reduce the pressure challenge any further. This justification must be based on an engineering feasibility analysis and not solely risk-benefit tradeoffs.

piping from overpressurization due to inadvertently starting the charging pumps, RCS pumps, SI pumps, and pressurizer heaters. The evaluation of the design for the SCS relief valves is discussed in Section 5.2.2 of this report.

The SCS piping and valves from the RCS up to and including SI-653 and SI-654, are designed to ASME Section III, Class 1. The remainder of the SCS piping, components and valves, including relief valves, SI-179 and SI-189, are designed to ASME Section III, Class 2. The relief valves, SI-179 and SI-189, are located inside the containment. The reactor coolant discharged through the relief valves is collected in the holdup volume tank (HVT), which is a low collection point in the containment. The spillage from the HVT is collected in the IRWST, which provides the borated water for the SIS. This relief flow path provides the System 80+ design with the capability to preserve the RCS inventory and the SIS water source in the containment, and to avoid flooding of any safety-related equipment should relief valves be stuck open.

The SCS design meets the pressure relief requirements of BTP RSB 5-1 and is acceptable.

5.4.3.4 SCS Pump Protection

Each of the SCS pumps, as described in CESSAR-DC Section 5.4.7.2.2.E and in ABB-CE's response to RAI Q440.58, has a minimum recirculation line to protect the pump from a potential low flow or no flow operating condition. The miniflow lines are routed from the pump discharge back to the pump suction.

A locally operated manual valve, that is located in each miniflow line to allow pump maintenance, is locked open during all operating modes. A heat exchanger in each miniflow line removes pump heat in the event of a closed pump discharge path should an operator make a mistake.

Individual flow and pump inlet/outlet Pressure ~~temperature~~ instruments monitor the condition of each of the SCS pump trains. A readout for each of these instruments is in the main control room. In addition, the SCS flow has a low flow alarm located in the main control room. An alarm alerts the operator to low piping from overpressurization due to inadvertently starting the charging pumps, RCS pumps, SI pumps, and pressurizer heaters. The evaluation of the design for the SCS relief valves is discussed in Section 5.2.2 of this report.

The SCS piping and valves from the RCS up to and including SI-653 and SI-654, are designed to ASME Section III, Class 1. The remainder of the SCS piping, components and valves, including relief valves, SI-179 and SI-189, are designed to ASME Section III, Class 2. The relief valves, SI-179 and SI-189, are located inside the containment. The reactor coolant discharged through the relief valves is collected in the holdup volume tank (HVT), which is a low collection point in the containment. The spillage from the HVT is collected in the IRWST, which provides the borated water for the SIS. This relief flow path provides the System 80+ design with the capability to preserve the RCS inventory and the SIS water source in the containment, and to avoid flooding of any safety-related equipment should relief valves be stuck open.

The SCS design meets the pressure relief requirements of BTP RSB 5-1 and is acceptable.

5.4.3.4 SCS Pump Protection

Each of the SCS pumps, as described in CESSAR-DC Section 5.4.7.2.2.E and in ABB-CE's response to RAI Q440.58, has a minimum recirculation line to protect the pump from a potential low flow or no flow operating condition. The mini-flow lines are routed from the pump discharge back to the pump suction.

A locally operated manual valve, that is located in each miniflow line to allow pump maintenance, is locked open during all operating modes. A heat exchanger in each miniflow line removes pump heat in the event of a closed pump discharge path should an operator make a mistake.

Individual flow and pump inlet/outlet ~~temperature~~ pressure instruments monitor the condition of each of the SCS pump trains. A readout for each of these instruments is in the main control room. In addition, the SCS flow has a low flow alarm located in the main control room. An alarm alerts the operator to low

Since the SCS takes suction from the RCS hot leg, RCS pressure has no effect on SCS pump flow.

flow conditions that lead to a loss of shutdown cooling due to either a loss of adequate pump suction or the closure of a system valve.

In a letter of January 21, 1993 and in Amendment S to CESSAR-DC Sections 5.4.7.2.2.E and 5.4.7.4, ABB-CE indicated that the SCS pumps are required to operate in a range from the design point to runout conditions. These pumps are not required to operate at reduced flow, such as injection mode following a small break LOCA. For post-trip long term cooling, the SCS is provided only after the RCS level has been stabilized, and the pressure and temperature have been reduced to low pressures, ~~to avoid low flow operating conditions which may cause pump damage resulting from the flow instability phenomena~~. ABB-CE evaluated operating data for pumps with the similar mechanical seal design used in SCS pumps and indicated that these pumps could be operated at the range of the design flow to runout conditions for more than 12,000 hours without overhaul. ABB-CE also indicated that the SCS pumps will be inspected, tested, repaired and replaced to meet the ~~pump functional~~ requirements in accordance with the requirements of the ASME OM Code-1990. Based on its evaluation of the SCS design discussed in this ~~section~~, the staff concludes that ABB-CE has provided reasonable assurance for the operability of SCS pumps. On this basis, Open Item 5.4.3.4-1 is resolved.

The SCS design satisfies the pump protection requirements of BTP RSB 5-1; therefore, the SCS pump design is acceptable.

CE response of 1/20/92 specified ASME OM-1990, but CESSAR-DC ~~spec~~ 3.9.6 specifies ASME/ANSI-OM-1987. 1987 edition has been accepted by NRC and is the correct reference.

5.4.3.5 Shutdown and Low Power Operation Risk

The NRC staff has had increasing concern over the safety of operations during low power operations or periods of plant shutdown. The Diablo Canyon event of April 10, 1987, highlighted a particularly sensitive condition of the operation of a PWR with a reduced inventory in the reactor coolant system. After the NRC reviewed the event, the staff issued GL 88-17, "Loss of Decay Heat Removal," on October 17, 1988. The letter requested that licensees address numerous generic deficiencies to improve operational safety during operation at reduced reactor coolant inventory. This included deficiencies in procedures, hardware, and training in the areas of (1) prevention of accident initiation, (2) early mitigation of accidents, and (3) control of radioactive

OK ayO

In CESSAR-DC Table 6.1-1, ABB-CE indicates that Inconel 690 will be used in lieu of Inconel 600 as an ESF pressure-retaining material. Operating experience indicates that Inconel 600 is susceptible to cracking. ABB-CE has considered using alternate materials that resist stress corrosion cracking. ABB-CE will use Inconel 690 for all applications where Inconel 600 was to be used. The staff views the Inconel 690 alloy as the preferred nickel base alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant (RC) and most secondary water chemistries. DSER Open Item 6.1-2 is considered closed.

OK ayO

ABB-CE is proposing to use Types 304 and 316 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking when the oxygen content of the RC exceeds 0.010 ppm at temperatures above 93 °C (200 °F) during normal operations. During start-up and operation of the nuclear steam supply system (NSSS), these temperature and chemical conditions are maintained through specified chemistry control. ABB-CE has taken alternative mitigating approaches as allowed in RG 1.44, "Control of the Use of Sensitized Stainless Steel," thus providing reasonable assurance of the integrity of austenitic stainless steel components in contact with RC. DSER Open Item 6.1-3 is closed.

Note 1

The ferrite content limits for austenitic stainless steel castings and weld metal in the System 80+ design are broader than those in industry guidelines (Ref. 1) and staff guidance (Ref. 2). ABB-CE during a meeting with the staff on June 22, 1993, stated that it will limit the ferrite content of austenitic stainless steel castings to a maximum of 20 percent. ABB-CE in Amendment L to CESSAR-DC modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes that these lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance of components of these materials maintaining adequate fracture toughness for their 60-year life. DSER Open Item 6.1-4 is considered closed. ?

Note 1

20%

My notes on Δ ferrite say for ↑ castings:

Δ ferrite 5FN to 30FN for temp ≤ 500F
8FN to 20FN " " > 500F

This is what we said in SAR except for ESF where we said 8FN to 30FN. Probably because ESF doesn't exceed 500F. See attached pgs from SAR other materials other than Type 300-series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels (COL Action Item 6.1.1-2).

OK ay

To prevent halide-induced intergranular corrosion which could occur in an aqueous environment with significant quantities of dissolved oxygen, the COL applicant will add hydrazine to inhibit the flushing water (COL Action Item 6.1.1-3).

In welding ferritic steel, the COL applicant will follow the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Sectic III of the ASME Code (COL Action Item 6.1.1-4).

This is not in agreement with our response to 6.1.1-4 (see attached)

6.1.2 Organic Materials

OK ay

The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on ABB-CE having met the quality assurance requirements of Appendix B to 10 CFR Part 50, since the coating systems and their applications will meet the positions in RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and the quality assurance standards of American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." Also, the containment coating systems have been evaluated as to their suitability to withstand a postulated (DBA environment. The coating systems chosen by ABB-CE have been qualified under conditions which take into account the postulated DBA conditions.

References

1. Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991.

COL Action Item 6.1.1-4:

In welding ferritic steel, the COL applicant will follow the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Section III of the ASME Code (COL Action Item 6.1.1-4).

Response to COL AI 6.1.1-4:

ABB-CE has taken exception to Position C.2 of RG 1.50 in Open Items 6.1-1; 5.2.3-1 and 5.3.1-7. We feel that the exception applies equally to the COL. The response to the referenced open items is:

Regulatory Guides are not requirements, and, therefore, exceptions are allowed if suitably justified. ABB-CE's basis for taking exception to Position C.2 of RG 1.50 is report, WCAP-8578, Effect of preheat and Post Weld Heat Treat on Hydrogen Induced Cracking in Pressure Vessel Steels, Sept. 1975. That report documents exhaustive testing to substantiate that there are alternatives to the procedure in Position C.2 which are equally effective in providing reasonable assurance that components made from low alloy steels will not crack during fabrication and which minimize the possibility of subsequent cracking. The report presents three acceptable alternatives for achieving reasonable assurance of freedom from cracks or from development of cracks later on. ABB-CE believes it acceptable to choose either option based on specific welds. One of the options is Position C.2, and ABB-CE does utilize this option in specific cases.

Recommended COL Action Item 6.1.1-4:

In welding ferretic steel one of the options of report WCAP-8578, Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels, Sept. 1975 will be followed as an alternate to RG 1.50 Position C.2. Additional requirements of the ASME Code will also be followed.

All raw austenitic stainless steel, both wrought and cast, used to fabricate pressure retaining components of the engineered safety features, is supplied in the annealed condition as specified in the pertinent ASME Specification (i.e., 1900-2050°F for 0.5 to 1.0 hour per inch of thickness and rapidly cooled below 700°F). The time at temperature is determined by the size and type of component. Vendor fabrication procedures will be audited to ensure that unstabilized austenitic stainless steel with a carbon content greater than 0.03% is not exposed to the temperature range of 430°C to 820°C (800°F to 1500°F) other than during welding.

Duplex, austenitic stainless steels, containing greater than 5FN delta ferrite (weld metal, cast metal, weld deposit overlay), are not considered unstabilized, since these alloys do not sensitize; i.e., form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

Example of what we said in Chap 6

CF 8M CF 8	Cast stainless steels (delta ferrite controlled to 5FN to 30FN)
Type 308 Type 309 Type 312 Type 316	Singly and combined stainless steel weld filler metals (delta ferrite controlled to 5FN to 15FN deposited)

In duplex austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenite interface during exposure to temperatures ranging from 800 to 1500°F. This precipitate morphology precludes intergranular penetrations associated with sensitized Type 300 series stainless steels exposed to oxygenated or otherwise faulted environments.

6.1.1.1.3.2 Cleaning and Contamination Protection Procedures

Specific requirements for cleanliness and contamination protection are provided for NSSS components which provide contamination control during fabrication, shipment, and storage as recommended in Regulatory Guide 1.37.

Contamination of Type 300 series austenitic stainless steels by compounds that can alter the physical or metallurgical structure and/or properties of the material is avoided during all stages of fabrication. Painting of Type 300 series stainless steels is prohibited. Grinding will be performed with resin or rubber bonded aluminum oxide or silicon carbide wheels that have not previously been used on any materials other than Type 300 series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels.

Solution heat treatment is not performed on completed or partially-fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described below.

B. Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, has been conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of RCPB components. The ASTM standard A 708 (Strauss Test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A 708 is utilized as a go/no-go standard for acceptability.

As a result of the above tests, a relationship was established between the carbon content of 304 stainless steel and weld heat input. This relationship is used to avoid weld heat-affected-zone sensitization as described below.

C. Unstabilized Austenitic Stainless Steel

The unstabilized grades of austenitic stainless steels with carbon content of more tnan 0.03% used for components of the RCPB are 304 and 316. These materials are furnished in the solution annealed condition. Exposure of completed or partially-fabricated components to temperatures ranging from 800°F to 1500°F is prohibited.

Duplex, austenitic stainless steels containing more than 5FN delta ferrite (weld metal, cast metal, weld deposit overlay), are not considered unstabilized since these alloys do not sensitize, that is form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

Example of what we said in Cha 5

CF8M, CF8	Cast stainless steel (delta ferrite 5FN to 30FN, 8FN to 20FN for normal operating temperature above 500°F)
308, 309 312, 316	Singly and combined stainless steel weld filler metals (delta ferrite controlled to 5FN-15FN deposited)

other materials other than Type 300-series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels (COL Action Item 6.1.1-2).

To prevent halide-induced intergranular corrosion which could occur in an aqueous environment with significant quantities of dissolved oxygen, the COL applicant will add hydrazine to inhibit the flushing water (COL Action Item 6.1.1-3).

In welding ferritic steel, the COL applicant will follow the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Section III of the ASME Code (COL Action Item 6.1.1-4).

6.1.2 Organic Materials

Some type of qualifier is required (e.g., intent, goal, or objective)

The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on ABB-CE having met the quality assurance requirements of Appendix B to 10 CFR Part 50, since the coating systems and their applications ~~will meet~~ the positions in RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and the intent of the quality assurance standards of American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." Also, the containment coating systems have been evaluated as to their suitability to withstand a postulated (DBA environment. The coating systems chosen by ABB-CE have been qualified under conditions which take into account ~~the~~ postulated DBA conditions.

The intent of ANSI N101.2 is met by selecting coatings based on ASTM D-3842, "Standard Guide for Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants," ASTM D-3911 "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," and ASTM D-3843, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities." These ASTM Specifications represent the current industry technology and experience and are intended to replace the ANSI Standards referenced in ANSI N101.2.

References

1. Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991.

2. U.S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for (BWR) Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, January 1988.

3. C.W. Marschall, M.P. Landow, and G.M. Wilkowski, "Effect of Dynamic Strain Again on Fracture Resistance of Carbon Steels Operating at Light-Water-Reactor Temperatures," ASTM STP 1074, ASTM, Philadelphia, PA, 1990, pp. 339-360.

4. K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the ASME, Subgroup on Fatigue Strength, on December 5, 1988, in New York City, NY.) (Enclosure in letter, from John W. Craig of NRC to Edward Griffing of the Nuclear Management and Resources Council, Washington, D.C., dated July 2, 1991.)

5. M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1992, pp. 293-306.

6. J.B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203.

6.2 Containment Systems

The containment systems for the System 80+ design include: (1) a containment structure as the primary containment; (2) a secondary containment (shield building) surrounding the primary containment ~~which houses equipment essential to safe shutdown of the reactor as well as fuel storage facilities~~; and (3) supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity to the environment with a design leakage rate of 0.5 percent of free volume for the first day and 0.25 percent of free volume each day after the first 24 hours. These values are predicated on a containment pressure caused by the DBA equal to the containment's design value of 365.4 kPa (53 psig) for the first 24 hours and equal to 50 percent of its design value (i.e., 182.7 kPa, or 26.5 psig) after 24 hours. The secondary containment is discussed in Section 6.2.3 (below).

6.2.1 Primary Containment Functional Design

The System 80+ primary containment design consists of a 61m diameter (200-ft) spherical steel shell with a nominal wall thickness of 4.45 cm (1.75 in.). This wall will be thicker around primary containment penetrations to structurally compensate for these openings. The primary containment will enclose the NSSS (i.e., reactor vessel, steam generators, RC pumps, pressurizer, and associated connecting piping), the in-containment refueling water storage tank (IRWST), safety injection tanks, the refueling canal, and associated mechanical, electrical, and heating, ventilation, and air conditioning (HVAC) support components.

The primary containment shell will be supported by embedding a lower segment between the containment internal structures concrete and the reactor building (RB) subsphere concrete. There is no structural connection between the free standing portion of containment and the adjacent structures other than penetrations and their supports. Thus, the portion of the spherical primary containment shell above the support region (elevation 91 + 9) will be structurally independent.

The primary containment will have a net free volume of 95,630 m^3 (3,377,000 ft^3) and is designed to withstand pressures and temperatures resulting from a spectrum of primary coolant and steamline pipe breaks and from a negative differential pressure caused by an inadvertent actuation of the containment spray system. The primary containment design parameters are an internal design pressure of 365.4 kPa (53.0 psig), an external design pressure of -13.8 kPa (-2.0 psig), and a design temperature of 143 °C (290 °F).

A comparison of the System 80+ containment design features with those of ABB-CE's other designs is presented in Table 6.2.1 of this SER. These features include containment structure type, power level, containment free volume, design pressures, design temperatures, and calculated peak DBA containment pressures and temperatures.

some.

A number of insights and conclusions can be drawn from Table 6.2.1. The System 80+ containment design represents a significant change from ~~all~~ of ABB-CE's previous containment designs, which consisted of steel-lined, pre-stressed, post-tensioned concrete structures with no secondary containment.

The System 80+ free volume will be considerably larger than all of ABB-CE's previous containments. Although this larger volume is expected when comparing the System 80+ design to ABB-CE's lower-power plants, the comparative ratios of containment free volume to power show that the System 80+ containment design will have a considerably larger free volume to power ratio than is found in ABB-CE's other designs. This ratio is between 19.2 and 19.9 m^3/MW (677 and 704 ft^3/MW) for ABB-CE's earlier designs, but is 24.6m^3/MW (869 ft^3/MW) for the System 80+. This larger relative volume is proportional to the relatively lower design pressure for the System 80+ (365.4 kPa, or 53 psig) as compared to ABB-CE's ~~other~~ designs (372.3 to 413.7 kPa or 54 to 60 psig).

Insert B

Table 6.2.1 also shows that the external design pressure of the System 80+ plant (13.8 kPa or 2 psig) is less than that of the System 80 and 3400 - megawatt thermal (MWt) CE designs (34.4 kPa or 5 psig). The System 80+ containment has a significantly higher design leak rate for the first 24 hours than do ~~all of ABB-CE's~~ other designs ~~except for the 3700 MWt design~~. This design leak rate is 0.5 percent per day, while the System 80 and 3400 MWt designs have a design leak rate of 0.1 percent per day. ~~The System 80+ margins between the calculated and design internal and external containment pressures are smaller than those for ABB-CE's operating plants~~. Containment design pressure margin will be discussed in Section 6.2.1.1.2 of this SER.

Insert A

Insert A

This higher design leak rate is a benefit to a utility because it provides more operating flexibility for containment leak rate testing and associated maintenance. This is consistent with the ALWR URD requirement. The 10CFR100 limits are still met with this higher leak rate.

Insert B

steel lined, pre-stressed, post-tensioned concrete containment

6.2.1.1 Containment Pressure and Temperature Response to High-Energy Line Breaks

The staff reviewed the temperature and pressure response of the primary containment to a spectrum of LOCAs and main steamline breaks (MSLBs), an analysis of negative or external pressure for the primary containment, and an analysis of the minimum containment backpressure for LOCA analyses. The staff will address the response of the secondary containment in Section 6.2.3 of this SER.

6.2.1.1.1 Containment Analytical Model

ABB-CE calculated the pressure and temperature response of the containment using the CONTRANS computer code. CONTRANS models both active and passive heat sinks in the containment and the energy source using information from a table of mass and energy releases. These heat sinks include containment spray and fan cooler and the materials inside the containment that can absorb energy ~~by convection~~ from the containment atmosphere.

The containment is simulated as a vapor and liquid region with time incremental step thermodynamic properties solved by CONTRANS. Mass and energy can be transferred between these two regions by boiling, condensation, and evaporation. CONTRANS models noncondensible gases and allows for a different temperature in each region. Liquid condensed from the atmosphere is automatically deposited in the sump or liquid region of the model.

The staff finds all but one of the analytical models and all of the assumptions used by ABB-CE acceptable to calculate the containment pressure and temperature transients following a LOCA and an MSLB in the System 80+ containment. The staff will address the exception concerning post-LOCA mass and energy release data in Section 6.2.1.3 of this SER.

6.2.1.1.2 Containment Pressure Response

The maximum differential pressure in the primary containment occurs from an MSLB at 0 percent power with the loss of one containment spray system. The

In addition, ABB-CE stated that a pipe break in these compartments would result in a negligible increase in pressure due to adequate venting. The staff concurs with ABB-CE's assessment and, therefore, DSER Open Item 6.2.1.2-1 is closed.

6.2.1.3 Postulated LOCA Mass and Energy Release

ABB-CE calculated the mass and energy release data for five LOCA breaks using the CEFLASH4A and FLOOD-MOD2 computer codes for the blowdown and reflood phases, respectively. These five LOCA pipe breaks include examples in both the hot leg and the cold leg and examples for which the safety injection flow is at minimum and maximum rates.

The staff previously approved both CEFLASH4A and FLOOD-MOD2, which ABB-CE has used extensively. Moreover, the staff cited the CEFLASH4A code in SRP Section 6.2.1.3 for the calculation of mass and energy release data. These codes model the primary system for mass and energy release calculations by solving equations for the conservation of mass, energy, and momentum. The calculations include the following heat sources: primary and secondary coolant; metal within the primary and secondary coolant systems; core power transient and decay heat, water from the safety injection system (SIS), and steam generator heat transfer. However, ABB-CE stated in CESSAR-DC Section 6.2.1.3.8 that it did not include energy from the metal-water chemical reaction. ABB-CE estimated the amount of this energy to be small and to have a very small effect on the containment pressures. The staff found that not using the metal-water energy is non-conservative and inconsistent with the requirements in GDC 50. Although the effect may be small, the staff stated in the DSER that ABB-CE should account for it because the containment pressure only had a margin of about one percent (based on the original 49 psig design pressure), as discussed in SER Section 6.2.1.1.2. Including this small energy source should not cause any difficulty but will resolve the concern of non-conservatism. Therefore, the staff determined that the metal-water energy should be included as an energy source. This was identified as DSER Open Item 6.2.1.3-1. In Amendment N, ABB-CE stated that considering the increased energy from the metal-water reaction for a nominal core power of 3914 MWt, the calculated peak containment pressure would only increase to 46.7 psig (322 kPa). This is below the containment design

Therefore, the staff concludes that the methodology, assumptions, and initial conditions for the LOCA mass and energy release rate calculations comply with SRP Section 6.2.1.3 and provide for a suitably conservative analysis of the parameters for the containment pressure and temperature response analysis.

6.2.1.4 Main Steamline Break Mass and Energy Release

In analyzing the temperature and pressure response of containment, ABB-CE used the SGN-III computer code to calculate the MSLB mass and energy release rate. ABB-CE considered a spectrum of eight different breaks that included four different power levels (102, 50, 20, and 0 percent), two different break areas [0.81 and 0.42 m^2 (8.72 and 4.5 ft^2)] and two postulated failures (loss of one containment spray train and main steam isolation valve (MSIV) failure). ABB-CE made conservative assumptions on initial conditions such as power level, steam pressure, and fluid inventories. The assumed active failures are failure of either one containment spray train or one MSIV.

prior to isolation flow from the

SGN-III accounts for delayed closure of isolation valves, ~~carryover from~~ intact steam generators, initial pumped feedwater flow, and the later expansion of the feedwater inventory into the steam generator. The code models the secondary system by dividing it into nodes with an appropriate balance of mass within each node. SGN-III also solves the conservation of energy and momentum equations and explicitly accounts for subcooled and saturated fluid in the system. In Item II.2 of SRP Section 6.2.1.4, the staff cited SGN-III as an acceptable computer code for calculating the release of mass and energy from an MSLB.

Therefore, the staff concludes that the methods, inputs, and assumptions in CESSAR-DC Section 6.2.1.4 for calculating the MSLB mass and energy release data for containment pressure and temperature analyses comply with SRP Section 6.2.1.4.

6.2.1.5 Minimum Containment Pressure Analysis for ECCS Performance

In Appendix K to 10 CFR Part 50, SRP Section 6.2.1.1.A (Item II.h) and SRP Section 6.2.1.5, the staff requires that ABB-CE analyze the minimum contain-calculated pressure and temperature conditions from any LOCA with sufficient margin. The temperature and pressure profiles in the CESSAR-DC for the spectrum of LOCA and main steam line breaks are acceptable for use in equipment qualification.

qualified for post-accident operation.

6.2.2 Containment Heat Removal Systems

In CESSAR-DC Section 6.2.2, ABB-CE cites the containment spray system (CSS) as the only containment heat removal system. This system consists of two 100-percent capacity redundant and independent trains, each consisting of a heat exchanger, pump, spray headers and associated piping, instrumentation, and valves. The CSS is Safety Class 2, seismic Category I, and is designed to the ASME Boiler and Pressure Vessel Code Section III.

Responding to several staff questions on CSS pump behavior, ABB-CE detailed the calculation of minimum pump net positive suction head (NPSH) and design features that prevent debris from entering the pumps while ensuring an adequate water supply during a DBA LOCA. ABB-CE's method for calculating minimum CSS pump NPSH during a DBA LOCA conforms with RG 1.1 "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," and SRP Section 6.2.2. The calculated minimum CSS pump NPSH with all appropriate conservative assumptions is 63.7 kPa (21.3 ft) for the pumps at their runout flow of 25,000 L per min (6,500 gpm).

ABB-CE also discussed the design features of the IRWST and holdup volume tank (HVT) which will provide a sufficient water inventory during all phases of a DBA LOCA and prevent debris from entering the CSS pumps. The spillways and screens between the HVT and the IRWST and at the entrance to the suction piping to the CSS pumps prevent particles greater than 2.3 mm (0.09 in.) in diameter from entering the CSS. Using the methods in RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," ABB-CE calculated that no air would enter the CSS pump at the normal rate of pump flow and that a maximum of 2 percent of air would enter at the CSS pump runout flow. These analyses and design evaluations meet the requirements of RG 1.82 and SRP Section 6.2.2.

see next page

tion for the System 80+ design. The SIS injects borated water from the IRWST with capacity of 2.07×10^6 L (545,800 gal) while the System 80 refueling water storage tank has a capacity of 1.90×10^6 L (502,760 gal).

The System 80+ design includes two additional high-pressure safety injection (HPSI) pumps, and two fewer low-pressure safety injection (LPSI) pumps than are found in the System 80 design. In the response to DSER Open Item 6.3.1-1 regarding the SI flowrate at the low pressure range, ABB-CE indicated in a submittal of November 24, 1992, that the addition of two HPSI pumps provides the required amount of SI flow delivered at low pressure to compensate for the removal of two LPSI pumps. The emergency core cooling system (ECCS) design for System 80+ is consistent with the EPRI ALWR Utility Requirements Document (URD) requirement, Number 5.4.3.1.2, which states that separate high and low head pumps shall not be used for ECCS. For System 80+, the analysis in CESSAR-DC 6.3.3 has shown that the SIS design can meet the performance acceptance criteria specified in 10 CFR 50.46 for the ECCS. In addition, two shutdown cooling pumps can be manually actuated for injection at low pressure to mitigate consequences of beyond-design-basis-events. In the emergency operations guidelines (EOGs), ABB-CE adds step (c.v) in the Functional Recovery Guidelines (FRG) (page 11-132 in EOGs) to instruct operators to use SCS for RCS inventory control. Step (c.v) instructs operators to "align the SCS pumps, once RCS is depressurized, to inject borated water from the IRWST to DVIs," if the SIS fails to maintain RCS inventory. Based on the staff evaluation discussed in Section 15.3.7 for the LOCA analyses and ABB-CE's guidance to use the SCS in EOGs for injection mode to mitigate beyond-design-basis-events, the staff concludes that the SIS design is adequate to address the ECCS acceptance criteria of 10 CFR 50.46 and the RCS inventory control for accident mitigation. On this basis, DSER Open Item 6.3.1-1 is resolved.

ABB-CE considered the reliability of the system (such as the design to meet seismic Category I requirements) in the design of the SIS, and includes redundant subsystems (such as the design to meet single-failure criteria) to enhance the overall reliability of the SIS.

As described in Table 2.4.7-1 of the ITAAC, the IRWST has a minimum volume of 545,300 gallons to permit dilution of radionuclides from the core. In the context that the SER presents IRWST volume (for SIS), the volume that should be quoted for System 80+ is 495,000 gallons minimum water volume above SIS/CSS pump suction line penetrations. This was added to Table 6.8-1 in Amendment T. The corresponding volume is 469,260 gallons for System 80.

divisions (revised in Amendment Q)

6.3.3 Pump Protection Design Requirements

Power is through two divisions, each division provides power to two buses, each SI train powered by a separate bus.

The System 80+ design contains four independent active SIS trains. Each train consists of a SI pump and its associated valves. System reliability is achieved with two electrical buses, each bus supplying power to two SI pumps, associated valves, and supported systems. At least two trains of the injection system would still be activated if a single failure occurred in the power supply system. In CESSAR-DC Table 6.3.2-2, ABB-CE included a failure modes and effect analysis, which demonstrates that no single active or passive failure could prevent the SIS from fulfilling its short- and long-term functions. Since the design includes both the onsite electric power supply system and the offsite electric power supply system, either of which permits functioning of the SIS, the SIS design meets the requirements of GDC 17.

The four SI pumps will be of sufficient size so that one SI pump, together with the SITs, will provide sufficient injection flow to prevent the core from being uncovered for small breaks up to at least a ~~30.48~~ 21.59-cm (~~12~~ 8.5 in.)-diameter break size (DVI nozzle ~~injection lines~~). During a large-break LOCA, two SI pumps, together with the SITs, will provide the required injection flow to meet functional requirements. (see Amend T, 6.3.2.2.4)

Diameter for defining pump capacity is based not on DVI line size [illegible] DVI nozzle diameter

In response to RAI Q440.63, ABB-CE performed an analysis to determine the NPSH available from the IRWST to the SI pumps. ABB-CE performed this analysis with sufficient margin to meet the regulatory position stated in RG 1.1. ABB-CE committed to verify the required NPSH for SI pumps to be met during the pump procurement process, and included an item to verify available ~~and required~~ pump NPSHs in its inspections, tests, and acceptance criteria (ITAAC) program for the SIS system. ITAAC does not commit to verify required NPSH. Required NPSH is determined by vendor.

In response to the staff's question (RAI Q440.61) regarding the SI pump capability to operate for an extended period of time, ABB-CE stated that the SI pump design is similar to the design of the steam generator feedwater pump. ABB-CE evaluated feedwater pump operating data and indicated that those pumps could be operated without overhaul for more than 5 years. ABB-CE also stated that the pump will be inspected, and submitted a recommendation for parts that

response indicated pumps were similar in design to boiler feed pumps used in fossil plants, not SG FW pumps.

should be replaced periodically. The COL holder will perform the routine ISI defined in the TS to verify that the performance of the SI pump meets the functional requirements.

To prevent the SI pumps from runout because of low RCS pressure after a large-break LOCA, the maximum SI flow (RAI Q440.62) will be limited to an acceptable runout value when the SIS is set up during preoperational testing with the RCS at atmospheric pressure. The main control room (MCR) will include pressure indicators for the SI lines, flow meters for the SI pump discharge flow, and the associated alarms. These devices will alert the operator to the possible degradation of SIS pump performance. The instrumentation for monitoring the SIS pump performance will be operable with and without offsite power.

In response to the staff's questions (RAI Q440.71, 72, and 73) regarding the SI operation at low flow conditions as described in NRC Bulletin 88-04, ABB-CE stated that the arrangement of the SIS mini-flow recirculation lines will preclude pump dead-head operation resulting from pump to pump interactions as discussed in NRC Bulletin 88-04. The only cross-connection in the SIS arrangement ties two discharge (mini-flow) recirculation lines together downstream of an orifice and a check valve, creating a flow path to the vented IRWST. This piping arrangement will not result in the operation of one SI pump to cause other SI pumps to operate at a flow rate that is lower than the required minimum for pump protection. ABB-CE also confirmed with information from pump manufacturers and utilities, that the minimum recirculation flow previously established for SI pumps applies to the System 80+ design and is adequate to protect pumps at low flow conditions. The design minimum SI pump flow ranges from 322 to 397 lpm (85 to 105 gpm) (9.7 percent to 12 percent of best efficiency flow) ~~according to the size of the SI pumps, motors, and piping for the System 80+ design.~~ The staff agrees that the proposed SIS arrangement will avoid pump dead-head operation. However, the staff required that ABB-CE discuss design criteria for the minimum SI recirculation flow, and provide pump operating data or test results demonstrating that the SI pump would be operable for an extended time at the proposed flow range. ABB-CE should also demonstrate in completing the ITAAC that the mini-flow bypass has adequate capability. This was DSER Open Item 6.3.3-1.

see next page A

comment to pg 6-44

(A)

The statement indicates that SI pump miniflow varies from 85 to 105 gpm based on pump, motor, pipe size. This is not correct for the System 80+ SI pumps, a minimum miniflow of 85 gpm is specified to be acceptable. The maximum of 105 gpm represents the upper limit that is allowed to ensure that an excessive miniflow does not reduce SI delivery to the RCS.

Should be "full design flow" per EPRI 5.4.3.8

6.3.7 Testing

The SIS will be designed and installed to permit the SIS pumps to be tested at full-flow conditions with the reactor at power, which is consistent with the EPRI URD requirements. The COL applicant will perform preoperational testing of the SIS to verify that the performance of the system and components meets design criteria. The COL applicant will periodically test and inspect the SIS components and subsystems in accordance with the ASME Code, Section XI, which constitutes compliance with GDC 36, to ensure the SIS will operate properly in the event of an accident. The COL applicant will also test the SIS components, including pumps and automatic valves, to confirm acceptable performance of each active component in the SIS.

ABB-CE committed to demonstrate the operability of the SIS by subjecting all components to preoperational and periodic testing to be consistent with RG 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors"; RG 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurizer Water Reactors"; and GDC 37, "Testing of Emergency Core Cooling System." SIS testing is COL Action Item 6.3.7-1.

6.3.8 Conclusions

The SIS includes the piping, valves, pumps, instrumentation, and controls for transporting heat from the reactor core after a LOCA, including a pipe break, or safety valve opening in the RCS that could discharge a volume of coolant greater than that to be replaced by the normal makeup system. The staff reviewed the SIS, including piping and instrumentation diagrams, failure modes and effects analyses, and design specifications for essential components. The staff reviewed the CESSAR-DC design criteria and design bases for the SIS and the manner in which the design conforms to these criteria and bases.

The staff concludes that, the design of the System 80+ SIS is acceptable because it meets the requirements in GDC 5, 17, 19, 20, 36, and 37, and the guidelines in SRP Section 6.3. In Section 15.3.7, the staff documents its evaluation regarding the acceptance criteria of 10 CFR 50.46 and GDC 35.

ABB-CE has included a preoperational testing program in CESSAR-DC Section 14.2.12.1.22 and an ITAAC program that includes testing to (1) verify that the design SI flow and head meet SI pump functional requirements; (2) confirm that the required pump NPSH for the SI pumps are available; and (3) determine the SI runout flow to be used at low-pressure conditions. The staff reviewed the ITAAC for the SIS and finds that the Design Description and ITAAC for the SIS system include the appropriate design commitments which are to be verified and, therefore, determines that they are acceptable. On this basis, DSER Confirmatory Item 6.3.8-1 is resolved.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with SRP Section 6.4 to verify that they will conform to the acceptance criteria in the applicable regulations of 10 CFR Part 50. Specifically, the SRP acceptance criteria require the system design to meet: (1) GDC 4, "Environmental and Dynamic Effects Design Bases," regarding accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases; (2) GDC 19, "Control Room," regarding maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; (3) Three Mile Island (TMI) requirement 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems; and (4) TMI Action Plan Item III.D.3.4 (NUREG-0737) requirements as they relate to providing protection against the effects of release of toxic substances, either on or off the site.

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 SSCs to perform their required safety functions. This was identified as COL Action Item 6.4-1 in the DSER. Upon further review, the staff has basis documentation. The above verification was identified as a COL Action Item 6.4-3 in the DSER. Subsequently, in CESSAR-DC, ABB-CE stated that the COL applicant will verify that the control room habitability system is consistent with the licensing basis documentation. Therefore, COL Action Item 6.4-3 is resolved.

Based on the above, the staff concludes that the control room habitability systems meet the acceptance criteria of SRP Section 6.4 and are, therefore, acceptable pending ABB-CE revising CESSAR-DC Table 15A-10 in Amendment V to reflect "6000 cfm," not "4000 cfm" nominal, post-accident filtered rate. This is part of FSER Confirmatory Item 1.1-1.

6.5 Containment Spray System

The containment spray system (CSS) is a safety-grade system designed to reduce containment pressure and temperature following a LOCA or MSLB and to remove fission products from the containment atmosphere during a LOCA. The staff reviewed the design basis for the fission product removal function of the containment spray system to verify that it is consistent with the assumptions made in the accidents evaluation of CESSAR-DC Chapter 15. The acceptance for the fission product cleanup function of the containment spray system is based on meeting the relevant requirements and criteria in SRP Section 6.5.2 and GDC 41, 42, and 43.

or CSAS (added in Amendment U)

The CSS consists of two 100-percent capacity redundant and separated trains, each of which has one pump, one heat exchanger, and associated spray nozzles, spray headers, piping, valves, and instrumentation and controls. The CSS pumps start upon receiving a SIAS and the spray header isolation valves open upon receiving a containment spray actuation signal. The CSS pumps are designed to be functionally interchangeable with the SCS pumps to allow the CSS pumps to back up the SCS pumps during refueling when the CSS pumps are not needed. The CSS pumps and heat exchangers can be manually aligned to provide LTC of the IRWST during post-LOCA feed-and-bleed operations when the steam generators are not available to cool the RCS.

backup to the SCS for

added in Amend U

ABB-CE stated that the NPSH design requirement has been addressed in the revised CESSAR-DC Section 6.5.3.4 (Amendment N). CESSAR-DC Section 6.5.3.4 states, in part, that the IRWST is the suction source for both SI and CSS pumps during short term injection and long term cooling modes of post-accident operation. The minimum available NPSH for the SI and CSS pumps was determined based on the minimum water level in the IRWST during accident conditions. The SI and CSS pumps are located in the RB subsphere and are placed low enough below the minimum IRWST water level to assure adequate available NPSH. The minimum IRWST water level elevation was determined to be 23m (75.5 ft). The calculated available NPSH for the CSS pumps ranges from 7.3m (24 ft) at the design flow rate of 18,900 L/min (5000 gpm) to 6.5m (21.2 ft) at a pump runout flow of 6500 gpm. This exceeds the CSS pump required NPSH of 6.1m (20 ft) at runout flow. In addition, the reactor cavity will not be flooded during a LOCA as was assumed in determining the minimum water level of 23m (75.5 ft) in the IRWST. Because of this, an additional volume of water will raise the minimum water level in the IRWST by approximately 0.6m (2 ft), thereby increasing the available NPSH. ABB-CE also addressed the prevention of debris blocking the return water from entering of the IRWST in CESSAR-DC Section 6.8. Based on this review, the staff finds that there will be enough water in the IRWST to cover the minimum CSS pump NPSH. Therefore, DSER Open Item 6.5-2 is closed.

In CESSAR-DC Section 6.5.4, ABB-CE states that the COL applicant will conduct preoperational tests on the CSS to verify the operability of all pumps, valves, spray headers, spray nozzles, heat exchangers, and instrumentation. Before operating the plant, the COL applicant will also conduct a series of hydrostatic tests in accordance with the ASME Boiler and Pressure Vessel Code (COL Action Item 6.5-1).

division (previously changed in CESSAR-DC)

The CSS is designed to withstand the impact of floods, pipe breaks, missiles, pipe whip, water hammer, over-pressurization, fire, and a LOOP. Each CSS train has a separate power supply through an independent electrical bus and EDG. Physical separation is used throughout the system to preclude common-mode failures and enhance fire protection capabilities. CSS pipes are made of austenitic stainless steel, conform to ASME Boiler and Pressure Vessel Code, Section III, and are classified as seismic Category I piping system. All

Also the RCGVS can be directed to IRWST.

reactor drain tank (RDT) through one or both of the parallel isolation valves. The RCGVS can also remove non-condensable gases or steam from the reactor vessel upper head through a vent line to the RDT through a flow-restricting orifice and one or both of the parallel isolation valves. The RCGVS isolation valves are closed during normal reactor operation. During shutdown or transient conditions, the operator will follow operating procedures to manually actuate (open) the RCGVS valves from the MCR in order to vent the reactor vessel upper head or the pressurizer steam space if the operator determines that non-condensable gases have collected in the reactor vessel upper head or in the pressurizer steam space. The RCGVS will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19.

The parallel isolation valves are powered by a normal ac power source and an emergency ac power source. In CESSAR-DC Table 6.7-3, ABB-CE documented a failure modes and effects analysis (FMEA) demonstrating that the RCGVS will maintain a vent path after a single failure of either of the valves or of the power source. This demonstration satisfies the requirements of GDC 17 and 34.

? To satisfy the requirements for natural circulation cooldown specified in BTP reactor service building 5-1, the RCGVS can remove the steam bubble and depressurize the RCS for the shutdown cooling mode of operation. This is discussed further in Section 5.4.3-1 of this report.

The RCGVS is designed to seismic Category I requirements in compliance with RG 1.29.

The RCGVS is designed to permit periodic inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Codes. This constitutes compliance with GDC 36.

In response to the staff's request (RAI Q440.134), ABB-CE committed to include the RCGVS as a safety-grade system in the System 80+ TS. This commitment is acceptable. This was designated as DSER Confirmatory Item 6.7.1-1.

6.7.2 Rapid Depressurization System

OK we say this in 6.7.

The RDS is designed as a manually operated safety-grade system that removes steam or water from the pressurizer through two isolation valves in each of two parallel depressurization lines to the IRWST. The RDS is designed to mitigate the consequences of a beyond-design-basis event such as a total loss of normal and emergency feedwater (TLOFW) event.

The RDS valves are closed during normal operation. These valves are motor-operated and fail in the "as is" position. The valves are designed to be operated during a station blackout (SBO). Thus, the Class 1E dc busses supply electrical power to the motor operators (through dc to ac inverters). The FMEA documented in CESSAR-DC Table 6.7-3 demonstrates that an RDS bleed path can be established in the event of a single failure in the mechanical equipment or battery banks with total SBO. This design feature satisfies GDC 17 and 34.

The design-basis event for determining the size of the RDS bleed valves is a TLOFW event. ABB-CE performed the analysis using a realistic version of the CEFLASH-4AS(REM) code with assumed best estimate decay heat values. The CEFLASH-4AS(REM) is documented in CEN-420. The staff determined that use of the realistic version of the CEFLASH-4AS code is acceptable because the RDS is designed to mitigate accidents beyond the design basis. In the analysis of determination of the SDS valve size, ABB-CE did not credit letdown, charging, and pressurizer spray. In the accident scenario, ABB-CE assumed that the initial RCS power and secondary steam are generated at the rated output. The primary and secondary safety valves open at lift pressures of 17.2×10^3 kPa (2500 psia) and 8.27×10^3 kPa (1200 psia), respectively, and the RCPs trip 10 minutes after the event is initiated. In the DSER, the staff stated that ABB-CE should justify that a delay time of 10 minutes to trip RCPs is conservative in calculating the required size for the RDS valves. This was designated as DSER Open Item 6.7.2-1.

In the response to DSER Open Item 6.7.2-1, ABB-CE provided in a submittal of November 24, 1992, and CESSAR-DC 6.7.1.2.1.C.2 (Amendment U), a technical basis for operator action time of 10-minute. ABB-CE followed the guidance in

ANSI/ANS-51.1-1983, "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," and considered a TLOFW event as a plant condition 2 event. According to ANSI/ANS-58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," a plant condition 2 events requires at least 10 minutes from a TLOFW initiation until four RCPs are tripped. The staff has relied on the guidance in ANSI/ANS-58-1984 for approval of TS changes to the Haddam Neck Plant (an October 22, 1990 letter from A. Wang of NRC to E. Mroczka of CYAPC). Therefore, the staff concludes that the assumption of operator action time of 10-minute to trip 4 RCPs in sizing the RDS valve is acceptable. On this basis, DSER Open Item 6.7.2-1 is resolved.

ABB-CE analyzed two cases (in the response to RAI Q440.22): (1) a TLOFW event with one RDS bleed path available, two SI pumps operable, and immediate operator action to open the RDS bleed path after the primary safety valves (PSVs) open, and (2) a TLOFW event with both RDS bleed paths operable, four SI pumps operable, and an operator action delay of 30 minutes to open the RDS paths after the PSVs open. The analysis shows that case 2 is the worst case, which requires larger RDS bleed valves, each of 1.95×10^{-3} m^2 (0.021 ft^2), to meet the acceptance criterion. This criterion requires the minimum level of mixture water in the RC to remain two feet above the top of the core through the transient. Since the calculated size of the valve depends heavily on the computer codes used for analysis, ABB-CE thus must confirm the ~~capacity of the~~ RDS valve capacity in the ITAAC program.

In the response to DSER Open Item 6.7.2-2, ABB-CE included an item in the SDS ITAAC to verify the valve flow capacity. Shop tests measuring the steam flow through the RDS valve will be performed. As described in CESSAR-DC Section 6.7.4.1 (Amendment T), the CEFLASH-4AS(REM) will be used to model the performance of the RDS. The computer model of the RDS valves will be initialized to predict the same steam flow rate as the vendor data from shop testing of the RDS valves. The code will then be used in simulating a TLOFW event. To demonstrate the adequacy of the RDS design, the calculated results using the CEFLASH-4AS(REM) will show that the reactor vessel water remains two feet above the top of the core during the TLOFW event, which is the event used for determination of the RDS valve sizes. The staff finds that ABB-CE's tests and analysis are adequate for the verification of the RDS valve capacity. Therefore, the staff concludes that DSER Open Item 6.7.2-2 is resolved.

The RDS also performs an important function in mitigating a severe accident. During a core melt, the system would allow the RCS to be depressurized and reduce the possibility of a challenge to the containment, such as from direct containment heating.

In response to the staff's question (RAI Q440.19) on the operator delay time to open RDS valves for severe accidents, ABB-CE proposed allowing the operator delay time of 1.5 hours to open the RDS valves after the PSV lifts for accident mitigation. The severe accident for determining the operator delay time is a total SBO including the failure of the onsite alternate ac power source, the complete loss of secondary-side heat removal capability, and no SI flow. ABB-CE used Revision 16 of the computer code, MAAP3.B to model the RCS, the steam relief system and the IRWST. The results demonstrated that a delay of 1.5 hours by the operator will not violate the acceptance criterion that requires the RCS be depressurized from 17.2 x 10^3 kPa (2500 psia) to 1720 kPa (250 psia) before the reactor vessel fails. ABB-CE has confirmed that the 1720 kPa (250 psia) criterion precludes a direct containment heating challenge. Confirmatory Item 6.7.2-1 is therefore, resolved (see Section 19.2.3.3.3 for the staff's evaluation for the resolution of this confirmatory item).

In response to the staff's RAI Q440.20, ABB-CE stated that equipment procurement specifications will define the expected environmental parameters to satisfy the requirements for equipment survivability. However, ABB-CE did not submit specific information. The staff asked ABB-CE to submit a discussion of the specific environmental parameters (such as temperature, pressure, moisture and radiation) and to submit the guidance for meeting the equipment survivability requirements during a severe accident. ABB-CE has submitted an evaluation and demonstrated that the RDS would be available in responding to severe accident conditions. DSER Open Item 6.7.2-3 is, therefore, resolved. The staff's evaluation of the RDS valve operability during the severe accidents is discussed in Section 19.2.3.3.7 of this report.

dampers (6.8.2.2 of Amendment U)

Four IRWST safety valves will protect the IRWST from overpressure. These safety valves are of sufficient size to accommodate the consequences of a DBA steam release and RDV actuation if the IRWST is not cooled. ~~Four vacuum breakers~~ will prevent a vacuum from forming in the IRWST during safety injection actuation, containment spray actuation, and normal drain down from the IRWST to the refueling cavity. The low-volume containment purge subsystem will vent the IRWST during normal fill of the IRWST. The dampers will also

If one or more PSVs or RDVs actuate, steam or water will flow through the main discharge lines to distribution headers in the IRWST and into the sparger heads. High-pressure jets of steam will be injected into the IRWST, where it will be condensed and mixed with the IRWST water. The IRWST, in this case, will function similarly to the suppression pool in the boiling water reactor plants. However, in earlier versions of the CESSAR-DC, ABB-CE did not address the hydrodynamic loads to the IRWST and SRS. In the DSER, the staff stated that ABB-CE must submit additional information to address this concern. This was identified as DSER Open Item 6.8-1. In Amendment N of CESSAR-DC, ABB-CE provided the requested information in Section 6.7.6, "Hydrodynamic Loads on the Safety Depressurization System (SDS)," and Section 6.8.4, "Hydrodynamic Loads on the Incontainment Refueling Water Storage Tank." Based on its analyses, ABB-CE concluded that the loads on the SDS piping and IRWST are within the design capability of piping and supports for safety relief valve piping and the design capability of the IRWST structural elements. The staff concurs with this analysis and, therefore, DSER Open Item 6.8-1 is resolved.

The in-containment water storage system is Safety Class 2. Therefore, it will be manufactured, and tested in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section III.

The piping and instrumentation diagrams for the IRWST include temperature and level indication, but no pressure indication for the tank. Responding to the staff's RAI Q440.69, ABB-CE stated that the level instruments that indicate the IRWST level provide high and low level alarms in the control room. The diagrams reference instrumentation requirements in CESSAR-DC Section 7.4.1.3, which ABB-CE had not submitted.

response to 410.102a did not propose using containment pressure. IRWST is provided with pressure indication; see 6.8.3

and

pressure; Responding to Q410.102a, ABB-CE submitted a description of the level, ~~and~~ temperature indication, ~~a justification for using containment pressure for IRWST pressure~~, a revised drawing without the reference to Section 7.4.1.3 and, a new Section 6.8.3 to define the instrumentation. This completes the information required and, therefore, is acceptable.

Responding to Q440.69, ABB-CE stated that the IRWST is designed to meet SRP Section 6.2.2 and RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems." The staff concludes that the IRWST design and analysis meet the requirements of RG 1.82 and SRP Section 6.2.2 as discussed in Section 6.2.2.

Based on the above evaluation, the staff finds the in-containment water storage system acceptable in meeting the applicable guidance of SRP 5.4.11, SRP Section 6.2.2, and RG 1.82.

Table 6.3-1. Safety injection system equipment comparison

reverse

	~~System 80+~~ System 80	~~System 80~~ System 80+
High Pressure Safety Injection Pumps Nos.	2	4
Design Flow -L per min (gpm)	3090 (815)	3090 (815)
Design Head -m (ft)	869 (2,850)	869 (2,850)
Low Pressure Safety Injection Pumps Nos.	2	0
Design Flow -L per min (gpm)	15,900 (4,200)	-
Design Head -m (ft)	102 (335)	-
Safety Injection Tank Nos.	4	4
Design Pressure-kPa (psig)	4800 (700)	4800 (700)
Water Volume, Normal -m^3 (cft^3)	52.6 (1,858)	52.6 (1,858)
Refuel Water Tank	1	1
Minimum Water Volume -L (gal)	$1.90x10^6$ (502,760)	$2.07x10^6$ (545,800)
Safety Injection Discharge Lines	cold legs	DIV Nozzles on the Reactor Vessel

In the context of this table, "SIS Equipment" the volume to specify is the SIS useable ~~vo~~ volume. As added to Table 6.8-1 of Amendment T, the "Minimum Water Volume Above SIS/CSS Pump Suction Line penetrations" is 495,000 gallons for System 80+; the corresponding number is 469,260 gallons for System 80.

The design certification includes a description of the design process. The ITAAC that address software differ from other system ITAAC in that the acceptance criteria describe attributes of the process to be used to develop the software and specific attributes of the final software product.

ABB-CE submitted a software development plan (SDP), NPX80-SQP-0101.0, "Nuplex 80+ Software Program Manual," which describes the complete software development process including hardware integration. This plan has been referenced in the SAR.

The staff reviewed ABB-CE's software development process based on guidance and criteria in the following standards:

1. American Nuclear Society (ANS) American National Standards Institute (ANSI)/Institute of Electrical and Electronics Engineers (IEEE) Std 7-4.3.2-1992, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations"

2. ANSI/IEEE Std 730-1989, "IEEE Standard for Software Quality Assurance Plans"

3. ANSI/IEEE Std 828-1983, "IEEE Standard for Software Configuration Management Plans"

4. ANSI/IEEE Std 1012-1986, "IEEE Standard for Software Verification and Validation"

5. ANSI/IEEE Std 1042-1987, "IEEE Guide to Software Configuration Management"

6. ANSI/IEEE Std 1058.1-1987, "IEEE Standard for Software Project Management Plans"

7. American Society of Mechanical Engineers (ASME) NQA-2a-1990 Addenda, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications"

"Software Safety Plans." ABB-CE's SSP addresses the topics described in IEEE P1228. ABB-CE's SSP addresses the topics presented in IEEE P1228 through references to the SDP, and references organizational structure and responsibilities, resources, methods of accomplishment, depth of effort, and integration of system safety with other program engineering and management activities. The staff concludes that the SSP is acceptable.

7.1.4.8 Software Development Process

The process for developing and implementing software complies with the regulatory requirements and industry standards governing those activities. The process meets both the requirements of 10 CFR Part 50 for I&C systems and the requirements in the design certification material. The safety-related SQA program will be implemented by the vendor and evaluated by the NRC in accord with 10 CFR Part 50 (Appendix B).

ABB-CE's SDP referenced in the SAR describes software development process and audit activities, which include the plans described in the previous sections. The staff reviewed ABB-CE's SDP using the standards listed above and confirmed that the SDP acceptably addresses the planning stage processes shown in Figure 7.1. The COL applicant will document its actions in the remaining phases of the software development process. The staff will review these documents during the ITAAC process audits as the software systems are developed and implemented.

The computer system development process shown in Figure 7.1 illustrates the relationship between the implementation of the software-based systems and the NRC staff's ITAAC audits.

The COL applicant will submit the documents listed in Figure 7.1 for staff audit through each life cycle stage of the development process. Following each audit, the NRC staff will issue a report documenting the completion of that ITAAC stage (inspection, test, and analyses) and list open items that will require resolution. ~~Significant unresolved items could prevent the NRC staff from confirming completion of the ITAACs and, therefore, must be resolved before the COL applicant proceeds to the next design stage.~~ substitute insert next page

INSERT 7.1.4.8

Significant unresolved items could prevent the NRC staff from confirming completion of the ITAAC~~s~~. Resolution of these items may result in changes to information in software design documents and materials that have been the basis for certain ITAAC confirmations made during previous NRC audits or conformance reviews. These ITAAC~~s~~ may require reconfirmation after the documents and materials are changed.

the Supply of Commercial Digital Computer Hardware and Software To Be Used in Nuplex 80+ Safety Systems"). This document is referenced in ABB-CE's SDP. ABB-CE will procure ~~the~~ commercial equipment used in safety-related ~~equipment~~ applications through a commercial dedication process that the staff reviewed and found acceptable in accord with the guidance in Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Application (NCIG-07)."

The commercial-grade computer hardware and software items will be incorporated in safety-related systems through a process that requires:

1. requirements for supplier design control, configuration management, problem reporting, and change controls

2. reviews of product performance

3. receipt acceptance of the commercial-grade item from the supplier

4. final acceptance, based on equipment qualification and software validation in the integrated system

These requirements have been placed into the system ITAACs. Any changes to these commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The first aspect of commercial dedication is the use of well-developed operating systems in plant-specific digital systems. ABB-CE states that proven technology will be employed in the design and development of the System 80+ I&C systems. ABB-CE's criteria for component service is that the component has been in service for at least 3,000 operating years and in the field at least one calendar year. The staff finds this requirement acceptable based on similar requirements in the EPRI Utility Requirements Document (URD) for ALWRs. The stated goal, to which both the staff and ABB-CE agrees, is to use the best available technology without using unproven designs.

4/9/94

7.2 Reactor Protective System

7.2.1 Discussion

The PPS consists of the equipment necessary to monitor selected plant conditions and to actuate a reactor trip and ESF components. The two major subsystems of the PPS are the RPS and the ESFAS. The RPS is described in this section. The ESFAS is described in Section 7.3.

The RPS will rapidly shut down the reactor when certain plant conditions approach safety system set points. The RPS is segregated into four completely independent channels consisting of sensors and transmitters, signal conditioning, bistable logic, core protection calculators (CPCs), local coincidence logic (LCL), and initiating relays. Safety-related, independent sensors in each channel will be continuously monitored to provide signals that are processed by the PPS in the following sequence of functions:

1. An auxiliary process cabinet (APC) in each channel receives that channel's sensor signals and converts the ~~4-20 mA~~ sensor signals to 0-10 V (analog).

2. The APC transmits the converted analog signals by ~~hardwired data link~~ discrete conductors to two redundant bistable programmable logic computers (PLCs) in the same channel, or to two redundant CPCs (32-bit minicomputers) in the same channel, depending on the signals. INSERT NEXT PAGE

3. The bistable ~~PLCs~~ processors first convert the analog input signals to discrete digital signals and then compare the signals to the appropriate setpoints. The resulting signal is a binary value: trip or no trip. The bistable ~~PLCs~~ processors also transmit the digitized signals to the discrete indication and alarm system (DIAS) and the data processing system (DPS) by fiber-optic data link for processing and display. In the initial CESSAR-DC, ABB-CE did not describe the hardware and software design of the bistable ~~PLCs~~ processors in sufficient detail (DSER Open Item 7.2.1-1). ABB-CE later submitted the following discussion.

4/4/94

Insert 7.2.1 Item #2

....one or both bistable processors and /or the CPC within the same channel. The bistable processors utilize Programmable Logic Controller (PLC) technology. The CPC uses 32 bit minicomputer technology.

4/4/94

Two bistable processors

~~Redundant PLCs~~ in each channel add reliability to each channel bistable function. Each bistable processor is assigned process measurements for comparison based on an analysis of transients in relation to the mitigating process. When multiple process measurements are available for monitoring a transient, they are assigned to different bistable processors. The design integrates various system components, features, and functions into a microprocessor-based unit that performs the entire trip function.

As stated above, analog input signals are directed to analog input modules (AIMs) within the bistable processors ~~PLC~~, where the signals are converted from analog form to digital form. The AIMs include self-test and automatic calibration features to eliminate the need for periodic calibration of inputs. Calibration of each bistable input is performed against a precision reference voltage source contained in each module. This reference voltage source requires an annual calibration or as required by the manufacturer. Drift between calibrations is detected by comparing across channels. Common mode drift is detected within the DPS by comparing validated values from the DIAS for postaccident monitoring (DIAS-P) and DIAS for normal monitoring (DIAS-N) systems.

Digitized analog values are automatically reported to the bistable central processing unit (CPU) during each PLC scan cycle. Within the bistable CPU, a comparator algorithm determines the pretrip and trip output states. Each output state is determined by comparing the digitized process (from the analog-digital (A/D) converter) to the setpoint (pretrip and trip) from the setpoint algorithm.

The PPS software is deterministic (i.e., repetitive and non-interrupt driven ~~with no interrupts~~) to ensure predictable system performance and response under all conditions.

Software is divided into two major categories: operating system software and application software. Operating system software consists of the PLC processor operating system, input/output (I/O) handling, communications,

7/4/94

The RPC prevents a DNBR or LPD trip. The TLCs use the most conservative insertion penalty factor in the DNBR and LPD trip algorithms. If the RPC does not result in sufficient thermal margin, a DNBR trip or a LPD trip, or both will be generated.

If a CEAC is out of service, the TLCs will use the available CEAC penalty factors to generate the CMI, RPC, and reactor trip signals. The CEDMCS also initiates CMI and RPC signals when a CEACs is out of service.

7. The results of the CPC calculations are compared to the corresponding set points for low DNBR and high LPD. The results of the comparisons are binary values, trip or no trip, which are transmitted to the channel's ~~redundant~~ LCL PLCs (two per channel).

8. The signals sent to the other channel redundant LCLs are transmitted by fiber-optic data link to ensure signal isolation.

9. A channel parameter may be bypassed for testing or maintenance from the local operator's module in the channel's PPS cabinet or from the PPS channel operator's module in the MCR. The bypass information is sent by fiber-optic data link to the interface and test processors (ITPs) in the four PPS channels. The ITP receives the bypass signal from the MCR or the channel PPS cabinet and receives the bypass status from the other three channels through fiber-optic data link.

10. The LCL PLCs in the four channels use the bypassed parameter status information to process the signals from the remaining channels in a two-out-of-four logic for no channels bypassed or a two-out-of-three logic for one channel bypassed. Once a parameter is bypassed, the interlock logic in the LCLs prevents another channel from also bypassing that signal. If the same parameter in another channel must be taken out of service for maintenance, that channel parameter is placed in trip status, which places the remaining parameter in a one-out-of-two trip state

4/11/94

11. Upon detecting a two-out-of-four trip state (or two-out-of-three trip state when a channel is bypassed), the channel's ~~redundant~~ LCL PLCs send the resulting trip signals through the initiation relays by ~~hardwired~~ discrete conductors ~~data link~~ to the channel's dedicated undervoltage trip circuit breaker and shunt trip circuit breaker.

12. The appropriate trip signals also pass through hardwired data link to the ESF initiation relays, which transmit signals through fiber-optic data link to each of the ESF LCL PLCs that process the corresponding trip (e.g., CSAS, SIAS, MSIS).

13. The coincidence signals are used in actuating the RTSS or ESF-CCS.

14. An ITP in each channel of the RPS monitors, tests, and controls the operational state of the bypasses ~~RPS~~. The ITP also provides fiber-optically isolated RPS channel status and test results information to the DPS and the DIAS.

15. The RPS trips the reactor on the following conditions (the trip processor is in parentheses):

- variable overpower (PLC)
- high logarithmic power level (PLC)
- high local power density (CPC)
- low departure from nucleate boiling ratio (CPC)
- high pressurizer pressure (PLC)
- low pressurizer pressure (PLC)
- low steam generator water level (PLC)
- low steam generator pressure (PLC)
- high containment pressure (PLC)
- high steam generator water level (PLC)
- low reactor coolant flow (CPC)
- manual trip (hardwired)

16. The actuation logic for the RPS interfaces with the undervoltage and shunt trip relays in the RTSS. The CEDMs receive power through the RTSS

in two parallel lines from two independent, full-capacity, motor-generator (M-G) sets connected in parallel. Each line passes through two trip circuit breakers in series (each actuated by a separate initiation circuit), so that, although both sides of the branch lines must be deenergized to release the CEAs, each side of the line can be interrupted by two separate means. The reactor trip circuit breakers trip the reactor if signals from Channel A or Channel C coincide with those from Channel B or Channel D (two-out-of-four coincidence). When the trip circuit breakers are tripped open, power is interrupted to the CEDMs, thereby releasing the CEAs to drop into the reactor core and shut down the reactor.

An operator can trip the reactor manually by actuating two adjacent switches in the MCR to interrupt the ac power to the CEDMs. Either of two independent sets of trip pushbuttons will cause a reactor trip, as will manual reactor trip switches at the reactor trip switchgear.

17. The non-Class 1E alternate protection system (APS) provides separate and diverse reactor trip and EFW ~~trip~~ actuation logic. The APS is described in Section 7.7.

Section 7.1 describes the acceptance criteria and guidelines for the RPS. Acceptance criteria for the review of the RPS are based on the system design meeting the relevant requirements of 10 CFR Part 50 and the applicable general design criteria (GDC) listed 10 CFR Part 50 Appendix A.

7.2.2 Conclusions

The RPS includes features that limit reactor fuel, fuel cladding, and coolant conditions to plant and fuel design limits. The system provides pre-trip alarms and trip actions that are within CESSAR-DC Chapter 15 analyses assumptions. ABB-CE has committed to ensure that instrument inaccuracies, bistable trip times, CEA travel times, and circuit breaker trip times are considered in the design of the system. The staff reviewed the ITAAC to verify that the functional requirements are addressed acceptably in CESSAR-DC Chapter 15 analyses. The staff finds the ITAAC to be acceptable.

7/4/94

System 80+ plant would be brought into safe cold shutdown conditions. The CMF analysis took credit for operator actions within 15-30 minutes of the event initiation.

To monitor the status of essential safety parameters, the DIAS-P displays and processors is hardwired directly from the APCs, except for the core exit temperatures, the reactor vessel level, and the subcooling margin monitor (SMM), which are obtained from the postaccident monitoring instrumentation (PAMI) computer. Figure 7.2 depicts this configuration. The DIAS-P and the back-up ESF systems include a set of displays and controls that are independent and diverse from safety-grade computer systems.

The design includes back-up controls dedicated for manually and independently actuating ESF systems through a dedicated link that bypasses all data links, network communications, and all computers with large software applications. Switches on the main control panel (MCP) in the MCR enable operators to actuate two trains of SI and one train each of containment spray, EFW, closure of mainsteam isolation valves, closure of containment air purge valves, and closure of a letdown isolation valve. Figure 7.3 is a schematic of this configuration.

A control signal from each switch is directed to loop controllers (LCs) at the lowest level in the digital control path. The LCs are programmable logic controllers that send signals to switchgear in motor control centers and electrical distribution panels that control plant components. Under normal conditions, the LCs provide signals to plant components in response to digital signals received through the communication network. The dedicated manual signals actuate plant components by overriding the input data received from the network interface.

This override feature is reliable because the software in the LCs Typically requires less than 6 kilobytes of memory and is completely deterministic. ABB-CE committed to test the LCs that have the override logic to ensure that a CMF of the protection system software will not prevent the dedicated signals from actuating their associated ESF functions.

(i.e. repetitive and non-interrupt driven)

The control building consists of the MCR, the technical support center (TSC), the computer room, the electric switchgear rooms, offices, and mechanical support equipment areas.

Instruments for the air conditioning systems provide controls and indications of the temperature and radioactivity levels of the areas sensed. Early-warning ionization-type smoke detectors are located in the supply, return, and outside air ductwork serving the control room area ventilation system.
containment

Equipment in harsh environments will be qualified (as required) in accordance with the requirements of IEEE 323-1974, RG 1.89 (Rev. 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 includes temperatures from 43 °C (110 °F) to 204 °C (400 °F), a saturated/superheated steam/air mixture, radiation total integrated doses (TIDs) up to 4 x 10^5 Gy (4 x 10^7 rad) gamma and 2 x 10^6 Gy (2 x 10^8 rad) beta, and 4400 ~~440~~ ppm boric acid followed by a Ph of 7.0-8.5 after 4 hours using disodium phosphate. ABB-CE states that no new harsh environment equipment will be required for the ABB-CE 80+ design beyond that which has been previously qualified for the System 80 (~~Palo Verde~~) design. No digital equipment, such as remote field multiplexers (RFMs), will be located in the harsh environment. Since the portion of the sensor channels that will be subjected to the harsh environment consists of sensors and wire leads, this commitment is acceptable. An environmental qualification ITAAC will verify that ABB-CE meets this commitment.

The staff reviewed the identification of the RPS systems and components designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The staff concludes that ABB-CE has identified those systems and components consistent with the design bases for those systems. The identification of these systems satisfies this aspect of GDC 2 and 4.

The COL applicant will type test components, verify acceptable separation of sensors and channels, and confirm qualification of the cabling by the site resistance. The staff believed that, when in service, reactor trip breaker trip attachments could be jammed without racking out the breaker and thus without causing a trip or alarm. EPRI's URD Chapter 10 Section 8.3.4.3 specifies consideration of features in the design of reactor trip breakers or their enclosures which make it difficult to tamper with the breaker in a manner that would prevent the breaker from tripping. Although conformance to the URD requirements is not mandatory, the Commission has asked for information on an applicant's reasons for deviations from the URD. The staff concluded that ABB-CE should address this URD requirement. Tamper resistance of reactor trip breaker enclosures was an open item that required resolution for the staff to complete its review (DSER Open Item 7.2.2.23-1).

By Amendment J, ABB-CE added CESSAR-DC Section 8 to Appendix 13A, "Instrumentation and Control Features for Sabotage Resistance." This section describes the sabotage resistance. The PPS includes the reactor trip breakers in rooms to which access is controlled. Within each room, cabinets that contain safety-related equipment are locked and annunciate an alarm in the control room when entered. ABB-CE stated that the reactor trip switchgear cabinets are designed such that access to the internals of the reactor trip breaker is not possible without racking out the breaker, and that a reactor trip circuit breaker cannot be racked out without causing a ~~reactor~~ the breaker to trip. This design precludes interference with the reactor trip function by attempts to jam the reactor trip circuit breaker. DSER Open Item 7.2.2.23-1 is resolved.

SAR Section 7.2.1.1 did not classify the RPS as a vital system. The staff considers the RPS a vital system; therefore, as required by 10 CFR 73.55(c), access to all RPS components required to trip the reactor should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not needed if access to each vital area is otherwise controlled.) This was designated DSER Open Item 7.2.2.23-2. In a submittal dated April 30, 1992, ABB-CE amended CESSAR-DC Section 7.2.1.1 to designate the RPS as a vital system. Open Item 7.2.2.23-2 is resolved.

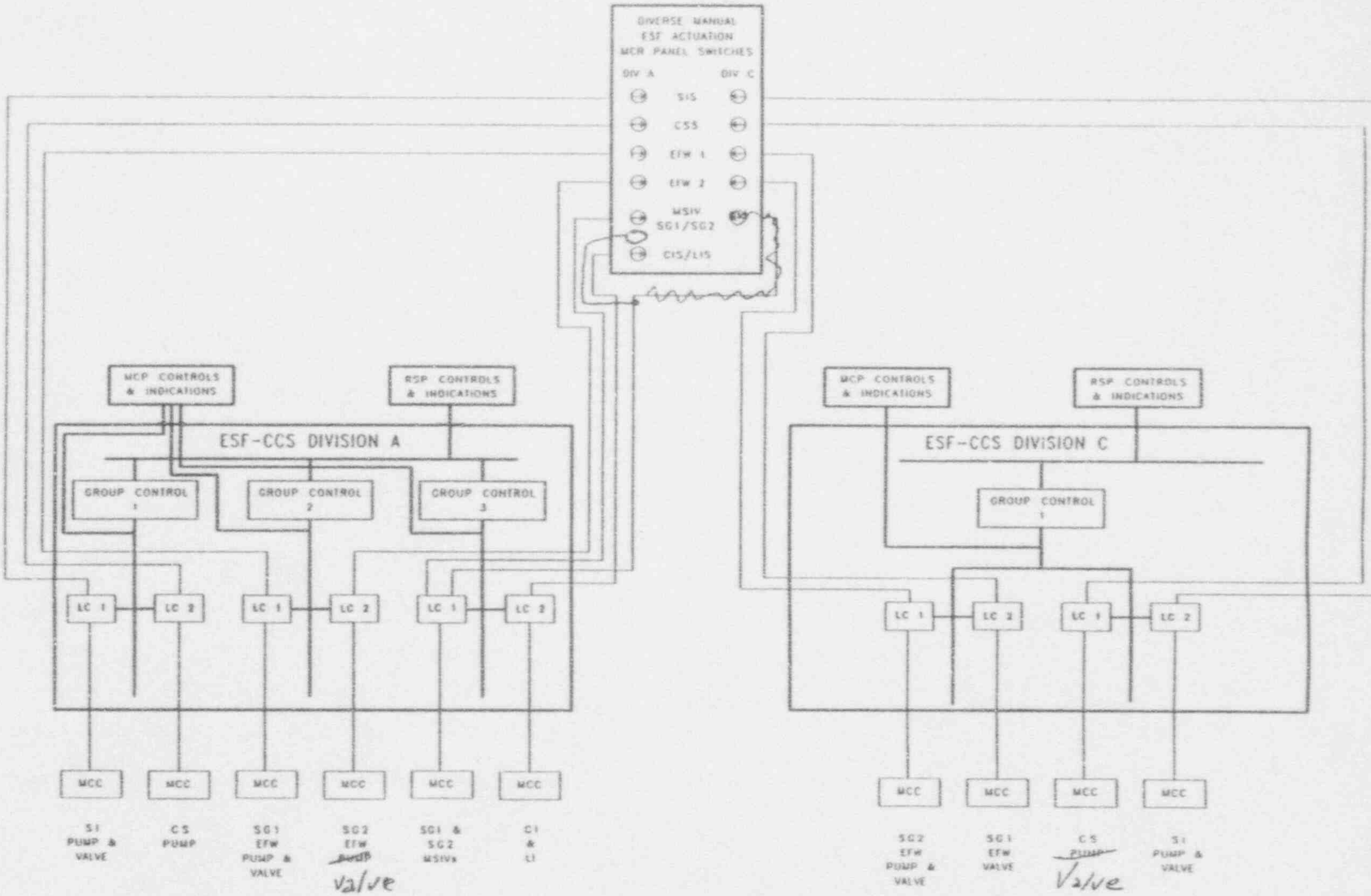


Figure 7.3. Diverse manual ESF actuation interface to ESF components

7.3 Engineered Safety Features Actuation System

7.3.1 Discussion

The ESFAS actuates plant systems to terminate or mitigate plant transient conditions after a plant upset. The ESFAS consists of the electrical and mechanical components for generating the signals that actuate the ESF systems, and that send the signals to the processors and equipment in each system in the ESF-CCS.

The bistable trip functions and coincidence logic in the PPS and component control logic (CCL) in the ESF-CCS generate the required component actuation signals. The control circuitry for the components generates the necessary signals to control the ESF-CCS. The ESFAS generates the following signals:

- containment isolation actuation signal (CIAS)
- containment spray actuation signal (CSAS)
- main steam isolation signal (MSIS)
- safety injection actuation signal (SIAS)
- emergency feedwater actuation signal (ESFAS)

LCL in the PPS consists of full two-out-of-four coincidence logic which sends actuation signals to the ESFAS initiation relays, which connect with the ESF-CCS selective two-out-of-four logic. These actuation signals generate the PPS permissive signals for automatic ESF component operations. Each signal in the actuation logic also sets a latch when the two-out-of-four coincidence is attained to ensure that the signal is not automatically reset once it has been initiated. This feature is acceptable.

The ESF-CCS consists of four independent divisions (A, B, C, and D), except for the CIAS and the MSIS, each of which consists of two independent divisions (A and B). Each division has primary and ~~secondary~~ standby processors. The primary processor controls the functions of the ESF-CCS. The ~~secondary~~ standby processor tracks the primary processor. If the primary processor fails, a redundancy controller automatically transfers control to the ~~secondary~~ standby processor if the secondary processor is available. This design meets the requirements for independence, redundancy, and reliability.

Each division consists of a ~~division~~ Division ~~master segment~~ Gateway and ~~multiple subgroup segments~~ Group Controllers (A1 through An). Each segment contains redundant PLC processors, local and remote multiplexers, and the necessary communications linkages. The ~~division~~ Division ~~master~~ Gateway segment supports the ~~links between the component control switches and the~~ master transfer switch operator's modules interfaces, the intersystem communication data links, and a maintenance and test panel. Each ~~subgroup~~ Group Control segment supports component control and data acquisition ~~links~~ interfaces and ~~related process controllers~~ component control switches.

Local and remote multiplexing is incorporated in the ESF-CCS to reduce plant wiring requirements. Remote multiplexers are located in the MCPs, the RSP, and the remote multiplexer cabinets near plant components and instrumentation. Fiber-optic cable provides isolation where it is required to meet the channel independence criteria of IEEE 279-1971, IEEE 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and IEEE 384-1977, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." This design for independence is acceptable.

Local and remote multiplexing can enhance data communications capabilities, although multiplexers can make the system more vulnerable to common-mode multiplexer failures. ABB-CE ~~must~~ has adequately addressed multiplexer vulnerabilities. ~~by (1) evaluating high-quality multiplexers that meet the requirements of industrial standards, (2) assessing vulnerabilities to CMFs, and (3) ensuring adequate defense in depth by considering the use of hardwired essential systems controls that bypass the multiplexer networks~~. Section 7.1 discusses CMFs and defense in depth. ABB-CE will procure the safety-related multiplexers through a commercial dedication process that has been reviewed by the staff and has been found to be acceptable as discussed in Section 7.1.

Commercial-grade computer hardware and software will be used in the safety-related systems through a process that requires:

1. requirements for supplier design control, configuration management, problem reporting, and change controls

2. reviews of product performance

3. receipt acceptance of the commercial-grade item from the supplier

4. final acceptance, based on equipment qualification and software validation in the integrated system

These requirements have been placed into the system ITAACs. Any changes to these commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The staff finds ABB-CE's commitment to implement a commercial dedication process as part of the ESF-CCS ITAAC to be acceptable, and DSER Open Item 7.3.1-1 is resolved.

The data communications networks are multidrop networks. They use active redundant cabling to maintain multiplexer interface operability should a single cable become inoperable. Communications with each ESF-CCS division are controlled by an intradivision communication network (ICN), which includes redundant cables to maintain intersegment communications should a single cable become inoperable. Hardware failures are annunciated to minimize the time required to repair or replace the defective module. This redundancy is acceptable.

For component control, each measurement channel is separated from the other three measurement channels to physically and electrically separate the signals to the ESF control logic. Cabling is separated within the cabinets, and signals to non-Class 1E systems are fiber-optically isolated. Each channel is supplied from a separate 120-V vital ac distribution bus. The independence and isolation are acceptable.

ABB-CE committed to develop system software in accordance with RG 1.152 (which endorses ANSI IEEE/ANS-7-4.3.2). The staff verified the adequacy of the software development process while reviewing the ITAAC. Software issues are discussed in Section 7.1. Any changes to these commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The ESF-CCS includes system-level selective two-out-of-four logic for ESF actuations, ~~subgroup~~ group control logic (~~SCL~~ GCL), component control logic CCL, selective group test (SGT) logic, and diesel loading sequencer (DLS) logic.

The ~~SCL~~ GCL controls ~~subgroups~~ groups of components. ESF functions are assigned to individual subgroup segments within each ESF-CCS division. For example, the SIS and the emergency feedwater system (EFWS) for steam generator 1 (EFW-1) are assigned to ESF-CCS subgroup segment 1, the containment spray system (CSS) and EFW-2 are assigned to subgroup segment 2. Functional assignment limits the effect of a single segment failure to ESF functions in a division. This segmentation is acceptable.

Functional assignments are further segmented within each ESF-CCS subgroup segment. For example, SIS and EFW-1 components and instruments for SIAS and EFAS-1 are assigned to separate input modules within a multiplexer to limit the effect of a single multiplexer or module failure to selected ESF functions within a division. ESF system interfaces are also confined within subgroup segments to minimize reliance on the ICN for ESF operability. For example, SIAS initiation signals and SIS component and instrument interfaces are confined to subgroup segment 1. Failure of the ICN, therefore, would not affect SIS operation. The staff finds the independence and redundancy to be acceptable.

Process controllers are supported where required from ESF-CCS subgroup segments to enable the operator to control continuous process control functions (such as valve modulation and auto/manual mode selection). This design satisfies the criteria of IEEE 279-1971.

Each ESF-CCS equipment cabinet includes a maintenance and test panel. The operator observes this panel for ESF-CCS equipment status and uses it for ESF-CCS maintenance, testing, and diagnosis. Each panel includes a master transfer switch (MTS) for transferring control from the MCR to the RSP. Each component is preprogrammed to remain in its current state or go to a predetermined safe state when the MTS is actuated. The staff accepted ABB-CE's commitment in the DSER to install redundant MTSs near the MCR exits and at the RSP. This was a confirmatory item. The SAR did not include sufficient design information for the staff to determine the adequacy of the MTS design. The lack of detailed information of the MTSs was designated DSER Open Item 7.3.1-2. ABB-CE submitted a more detailed design description in response to this ~~open~~ now closed item. The staff reviewed the MTS design as discussed in Section 7.4.6.

Setpoint values are administratively controlled and automatically monitored continuously. The fixed setpoints are adjusted at the PPS cabinet. Access for setpoint adjustment is limited by keylock with the access annunciated by the DIAS. The setpoints may be displayed at the PPS cabinet and through the DPS cathode ray tube (CRT) displays in the MCR.

The components of each ESFAS are segregated into functional groups, such as a valve group and a pump group. The operator in the MCR performs ESFAS selective group testing. This testing overlaps the PPS automatic testing of the ESF-CCS selective two-out-of-four coincidence logic and includes complete testing of the ESFAS through to the actuation of the group components. Testing is conducted one group at a time, thus preventing the undesired actuation of an entire ESF system during testing. Since this testing causes components in a group to actuate, an ESFAS signal from the PPS will not be impeded and the ESF system will be fully actuated regardless of the testing

Auxiliary and supporting systems for the safety-related I&C system cause a system-level bypass indication when these systems are bypassed or deliberately made inoperable. A bypass indication is provided for the safety-related system affected by the bypass. This indication of bypassed systems satisfies IEEE 279-1971.

The bistable trip channel interlocks for ESFAS, located in the PPS, prevent the operator from bypassing more than one trip channel or a trip parameter at a time. Different trip parameters may be bypassed simultaneously, either in the same channel, or in different channels. If a complete channel is disabled such as loss of a vital power supply, the operator will manually bypass all trip bistables for that channel, thereby rendering that channel bypassed. This is a manual action from the operator's control panel.

The operators receive the bypass status on displays in the control room, and through displays on the DPS. This satisfies the guidance in RG 1.47.

During PPS testing, interlocks prevent disabling of more than one redundant protection function at a time and prevent maintenance personnel from inadvertently causing unwarranted ESFAS signals. The bypasses satisfy IEEE 279-1971.

7.3.5 Redundancy

Circuits in the four PPS channels initiate four independent channels for each parameter from the process sensor. Four redundant ESF-CCS divisions operate four (or two) completely redundant ESF trains. Redundant components of the engineered safety system are assigned to redundant ESF-CCS divisions to maintain the desired level of design redundancy.

Redundant flow paths in the system design ensure cooling capability under single-failure conditions. These flow paths are assigned to different divisions of the ESF-CCS to maintain flow availability under single-failure conditions within an ESF-CCS division.

A redundant flow path may contain~~s~~ two valves in series to preclude spurious flow initiation upon single failures. Each valve is assigned to an independent testing is subject to the staff's acceptance of the TS. This was designated DSER Open Item 7.3.7-1. Staff acceptance of the TS is discussed in Chapter 16. This item is now closed.

During reactor operation, the DIAS and the DPS continuously monitor the sensors to detect sensor failures or degraded sensor signals. Section 7.2 includes additional details regarding sensor signal validation and bistable trip testing. The DPS performs additional testing of the bistable functions for the process control setpoints and alarms.

The DPS continuously monitors set points and activates alarms upon finding excessive setpoint deviation between redundant channels. While testing selected groups, operators periodically verify PPS bistable trip accuracy and interlock performance by manually varying the digital interlocking parameter from the operators module in the MCR. Analog-to-digital conversion accuracy is also periodically verified at the operator's module during sensor testing. The overlap of testing enables operators to verify PPS bistable trip accuracy and interlock performance.

The PPS test function automatically tests the initiation logic for each engineered safety system to determine its ability to generate an initiation signal. Testing begins by interrogating the status of the input signals to the logic and the state of the output. The test function compares the value of the output signal with the value of the input signal. Discrepancies are annunciated and a message is provided to the operator to describe the error. If there is no discrepancy, the testing function continues. Based upon the known inputs, the testing function will generate combinations of input signals and monitor the outputs of the logic for correctness. The testing function cannot change a genuine coincidence signal.

Testing is performed one channel at a time to avoid inadvertently actuating equipment with the testing function. This testing is done with the ESFAS initiation relay testing.

The ESF-CCS actuation logic is a selective two-out-of-four circuit controlled by signals from the initiation relays from the four PPS channels. These of an ESF system. Since this testing causes ESF components to actuate, an ESFAS signal from the PPS will not be impeded, and the ESF will proceed to full actuation.

The diesel load sequencer may be tested while the DLS is on line. During normal operation, all output control signals are disabled, allowing all logic functions to be tested without disturbing plant equipment. The outputs become enabled automatically when a valid initiation signal is received. Consequently, testing may be conducted without impeding load sequencer operation in the event of a station black-out condition. There are three distinct test phases: automatic testing, input testing, and load shed testing. ABB-CE's testing processes satisfy the criteria of IEEE 279-1971.

7.3.8 Vital Instrument Power Supply

Chapter 8 discusses the vital instrument power supply design.

7.3.9 Actuated Systems

The ESF systems are maintained in a standby mode during normal plant operations. The ESFAS generates actuation signals to ensure that the ESF-CCS performs the required protective actions. In SAR Section 7.3.1.1.10 ~~7.3.9~~, ABB-CE briefly described the I&Cs of each ESF system. The staff reviewed the ESF system actuations for the design-basis events and the monitored parameters required for ESF systems actuations.

7.3.9.1 Containment Isolation System

The containment isolation system enables the CIAS to isolate fluid systems that pass through containment penetrations so that any radioactivity released into the containment after a DBA will be confined within the containment. A high containment pressure signal or a low pressurizer pressure signal sends a CIAS and sends a SIAS.

Valves that must be isolated following a CIAS are installed with either air-operated controllers or motor-operated controllers. Air-operated valves fail

The CHRS prevents the concentration of hydrogen in the containment from reaching flammability limits after a design-basis LOCA. The CHRS may be manually initiated upon indication of high hydrogen concentration. The DPS monitors hydrogen concentration and sends the information to the operators in the MCR and at the RSP for viewing on the DPS CRT displays. ABB-CE did not state in the DSER whether hydrogen concentration is provided as a Class 1E alarm. The staff initially determined that this system should be Class 1E as required in IEEE 279-1971 (DSER Open Item 7.3.9.6-1). Emergency procedures require the hydrogen recombiners to be connected and activated within ~~72~~ 84 hours of a LOCA, which is before reaching the high hydrogen concentration setpoint for design basis events. The hydrogen recombiners then operate continuously. Since the operator is not required to operate the hydrogen recombiners after they are actuated, (e.g., upon detecting a high hydrogen concentration) no operator action is required to perform this safety function after the recombiners have been actuated. Consequently, the staff concludes that the alarm on high containment hydrogen is not required to be a Class 1E alarm. This closes DSER Open Item 7.3.9.6-1.

7.3.10 Conclusions

At the time that the DSER was completed, the staff lacked sufficient design information to conclude that the ESFAS and ESF-CCS met the relevant requirements of GDC 2, 4, 20 through 24, 34, 35, 38, and 41 and IEEE 279-1971 (as required by 10 CFR 50.55(h)). The staff noted that a significant portion of the system design is software-based, and could not be reviewed. Subsequently, the staff evaluated software aspects of the system design as part of ABB-CE's ITAAC submittal. The staff verified that other portions of the ESFAS and ESF-CCS design meet the relevant criteria based on the following.

The staff reviewed the ESFAS and ESF-CCS for conformance to the applicable regulatory guidelines in Section 7 of NUREG-0800 (SRP) and found reasonable assurance that the systems will conform to the applicable guidelines.

The staff also determined which systems and components of the ESFAS and ESF-CCS are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. ABB-CE designed these systems

The staff reviewed the design of the ESFAS and plant operating control systems, found that no portion of the EFAS is used for both protective and control functions and signals sent from EFAS to other systems are isolated through fiber-optic cables for signal transmission. Therefore, the staff concludes that the ESFAS satisfies the requirements of IEEE 279-1971 with regard to control and protection system interactions. The staff concludes that the ESFAS satisfies the requirements of GDC 24.

The staff concludes that the ESFAS design satisfies the requirements of 10 CFR 50.55(h) and IEEE 279-1971.

The staff evaluated the ESF-CCS for conformance to the requirements for testability, operability with both onsite power and offsite power, and single failures consistent with the GDC applicable to these systems. The staff concludes that the ESF component control systems are independent and satisfy the single-failure criterion and, therefore, meet the relevant requirements of GDC 34, 35, 38, and 41.

The staff reviewed the dependence of the ESFAS and ESF-CCS on the availability of the essential auxiliary supporting systems and concludes that the designs of the ESFAS and ESF-CCS are compatible with the functional performance requirements of the supporting systems. The staff, therefore, concludes that the interfaces between the design of the ESFAS and ESF-CCS and the design of the essential auxiliary supporting systems are acceptable.

All portions of the ESFAS and ESF-CCS designs are acceptable. ~~except for~~ The detailed design of the digital I&C system~~, which is addressed~~ is controlled by the Software Program Manual and verified by the relevant ITAAC (Section 7.1). ABB-CE resolved the possibility of CMF in digital systems and acceptable diversity in digital systems. The staff reviewed ABB-CE's diversity analysis, and found the analysis to be acceptable.

ABB-CE addressed multiplexer vulnerabilities by (1) evaluating high-quality multiplexers that meet the requirements of industrial standards, (2) assessing vulnerabilities to CMFs, and (3) ensuring adequate defense-in-depth by

7.5.2.4 Control Element Assembly Position Indication

The pulse-counting CEA position indication system and the reed-switch CEA position-indication system are two diverse, independent CEA position-indication systems that give CEA position information to the operator.

The CEDMCS receives automatic CEA motion demand signals from the reactor-regulating system (RRS) or manual motion signals from the CEDMCS operator's module and converts these signals to dc pulses that are transmitte to the control element drive mechanism (CEDM) coils to cause CEA motion. The CEDMCS counts these pulses to infer the CEA position by electronically monitoring the mechanical actions within each CEDM to determine when a CEDM has raised or lowered the CEA. Section 7.7 discusses the pulse-counting system.

A series of magnetically actuated reed-switch position transmitters (RSPTs) in the reed-switch CEA position-indication system send signals representing CEA position. Two independent reed-switch assemblies monitor each CEA ~~group~~. The RSPTs send an analog position-indication signal and three physically separate discrete reed-switch position signals. The analog position-indication system includes a series of magnetically actuated reed switches spaced at 3.8-cm (1.5-in.) intervals along the RSPT assembly with precision resistors between the switches. These switches form a voltage divider network. The RSPT is adjacent to the CEA extension shaft and actuating magnet. The analog output signal is proportional to the CEA position within the reactor core.

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 containment 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 2×10^6 Gy (2×10^8 rad) beta, and ~~440~~ 4400 ppm boric acid followed by a pH of 7.0-8.5 after 4 hours using disodium phosphate. 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.

Instrument channels are electrically independent and physically separated from each other and from non-safety-related equipment by qualified isolation devices. The DIAS-P and DIAS-N displays give credited redundancy for the display of Category 1 variables. These displays are electrically independent and physically separated. The DPS also presents each Category 1 variable to avoid TS limitations for conditions when a DIAS channel is out of service.

The DPS is physically separated and independent of both DIAS channels. Independent Class 1E power busses supply power for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, A ~~C~~ and B ~~D~~ instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The Category 2 variables are displayed on DIAS-N and DPS with power supplies from the non-safety-related instrument buses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the X or Y instrument bus. The redundant information systems conform to the guidelines for the physical independence of electrical systems in RG 1.75.

The ranges of the Category 1 indicators conforms to the ranges specified in RG 1.97. Where the required range of the monitoring instruments results in a loss of sensitivity during normal operating conditions, ABB-CE will install a separate instrument channel. The sensor instrument spans and setpoints conform to the guidelines of RG 1.105.

The staff reviewed the systems for which a bypassed or inoperable status is indicated in the control room. The list of systems includes all of the ESF connections between redundant or diverse safety systems where such connections exist for testing or maintenance. These indications satisfy the requirements of 10 CFR 50.34(f)(2)(xi) (TMI Action Item II.D.3).

The staff reviewed these systems to confirm that such design considerations as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these systems and commensurate with the importance of the safety functions to be performed.

The concerns discussed in Sections 7.1, 7.2, and 7.3 apply to the digital computer systems used in interlock systems.

7.6.1 Shutdown Cooling System Suction Line Valve Interlocks

The SCS is a low-temperature, low-pressure system that removes decay heat from the RCS. The steam generators cool the RCS to approximately 177 °C (350 °F) and 2758 3103 kPa (400 450 psia). Below these values, the SCS cools the RCS to refueling temperatures and maintains these conditions for extended periods.

Redundant, motor-driven, interlocked isolation valves on each suction line prevent overpressurization by preventing the suction line isolation valves from being opened if RCS pressure has not decreased below an acceptable value. The pressurizer pressure safety-related instruments are used to display RCS pressure conditions.

The SCS interlocks are redundant so that any single failure will not cause a suction line and heat exchanger to be subjected to pressures greater than

7.6.2 Safety Injection Tank Isolation Valve Interlocks

The SIS will inject borated water into the RCS upon receiving an SIAS and will supply long-term cooling in conjunction with other systems after an accident. The SITs inject borated water into the RCS if system pressure decreases below the SIT internal pressure. During normal operation, the motor-operated isolation valve on each tank is open with power removed from its motor circuit to eliminate the possibility of spurious closure. As the RCS pressure is reduced during plant shutdown, the low pressurizer pressure trip setpoint is reduced to avoid inadvertent initiation of SI, the SITs are depressurized to a value below the SCS design pressure, and the valves have their power restored and are closed.

The SIT interlocks prevent the SITs from inadvertently pressurizing the SCS while maintaining the SITs available in case of a LOCA. The isolation valves are manually closed when RCS pressure decreases below ~~2963~~ kPa (475 ~~415~~ psig) so that the SITs cannot overpressurize the SCS while the SITs are maintained at some pressure above atmospheric. As RCS pressure increases, the SIT isolation valves will automatically reopen at ~~3549~~ Kpa (600 ~~500~~ psig) to ensure that SITs are available for injection during plant startup. If the isolation valves are closed and an SIAS is received, the isolation valves will automatically open. The SIAS overrides the interlock or any manual signal.

The ESF SIS meets the requirements of the GDC, and the guidelines of RGs, and IEEE standards appropriate for ESF systems. The SIS valve indications satisfy the requirements of 10 CFR 50.34(f)(2)(xi) and TMI Action Item II.D.3.

- pressurizer pressure control system (PPCS)
- pressurizer level control system
- megawatt demand setter
- feedwater control system (FWCS)
- steam bypass control system (SBCS)
- reactor power cutback system
- boron control system
- in-core instrumentation system
- ex-core neutron flux monitoring system
- boron dilution alarm system
- alternate protection system
- process component control system
- cavity flooding system (CFS)
- hydrogen mitigation system (HMS)
- advanced control complex (ACC)
- discrete indication and alarm system
- integrated process status overview
- nuclear steam supply system integrity monitoring system
- data processing system

SG tube rupture detection inst?
Reduced inventory inst?

The acceptance criteria for the review of these systems are GDC 13, "I&C," and GDC 19, "Control Room." These I&C systems use digital technology to meet the design requirements and, hence, the staff's concerns and positions in Sections 7.1, 7.2, and 7.3 apply also to these systems.

The FWCS dedicated to each steam generator automatically controls the steam generator downcomer water level during power operations between 5 percent and 100 percent. Steam generator level is controlled during the following conditions if all other control systems are operating in automatic mode:

- steady-state operations
- 1-percent-per-minute turbine load increases between 5-percent and 15-percent NSSS power and 5-percent-per-minute turbine load increases between 15-percent and 100-percent NSSS power
- 10-percent turbine load steps between 15-percent and 100-percent NSSS power
- loss of one of three ~~or two~~ operating feedwater pumps
- load rejections of any magnitude

Below 15-percent NSSS power, the FWCS dynamically compensates for the steam generator level signal by sending a flow demand signal to a downcomer valve program where a downcomer feedwater valve demand signal is generated. The signal, or a manual signal from the operator, is passed to the downcomer valve. The signal controls the valve position. When the FWCS is in this control mode, the economizer control valve is closed and the pump speed setpoint is set to near its minimum value.

As NSSS power increases above 15 percent, the downcomer valve closes and the economizer valve opens to regulate the feedwater flow into the steam generator. The steam generator level signal is compensated by the difference between the total feedwater flow rate and the total steam flow rate. The resulting signal is subtracted from the level setpoint signal and sent through a proportional plus integral (PI) controller to produce a total feedwater demand signal. This signal produces an economizer valve demand signal. The operator can also control this signal manually.

The signal is also used to produce a feed-forward demand signal for the feedwater pumps. The pump program generates a pump speed setpoint signal which is passed to one of the feedwater pumps. The operator can also control this signal manually.

~~The main feedwater system has two variable-speed motor-driven main feedwater pumps normally operating, and one variable-speed motor-driven pump that will operate if a main feedwater train is lost. Selector switches in the MCR bring the backup pump on line. An interlock prevents operating more than two of the feedwater trains simultaneously. The selector switches also permit any combination of two main feedwater pumps to be operated from one process controller if an FWCS channel fails.~~

The main feedwater system isolation valves (MFIVs) will close within 5 seconds after receipt of a MSIS, even if the effects of a single failure are imposed. The MSIS is actuated on high containment pressure, high steam generator level,

The FWCS has three 50% capacity motor-driven main feedwater pumps normally operating. The FWCS is designed to automatically control steam generator level during a loss of one of three operating feedwater pumps (excluding the startup feedwater pump).

7.7.1.7 Reactor Power Cutback System

The RRS, PPCS, PLCS, MDS, FWCS, and SBCS work together to control minor changes in power and flow expected during normal NSSS operation. However, the PCS includes the RPCS to maintain the NSSS within the control band ranges during certain large plant transients such as a large turbine load rejection, turbine trip, or loss of one of ~~two~~ three online main feedwater pumps. Under these conditions, the NSSS can be maintained within the control band ranges by reducing NSSS power more rapidly than by a normal high-speed CEA insertion. Rapidly reducing NSSS power will also increase the thermal margin to accommodate inward CEA deviations (including rod drops) without a reactor trip.

The RPCS accommodates certain imbalances by dropping one or more preselected groups of full-strength CEAs into the core to reduce power incrementally. The RPCS also sends control signals to the turbine to rebalance the turbine and reactor powers, and to restore steam generator water level and pressure to control values.

The RPCS receives signals indicating loss of any operating feedwater pump (two signals for each pump), two cutback demand signals from the SBCS, and four cutback demand signals from the CPCs (one signal from each CPC). A two-out-of-two logic actuates the system for load rejections or loss of a feedwater pump. A two-out-of-four logic actuates the system for CEA deviations to be consistent with the two-out-of-four trip initiations from the CPCs and the

7.7.1.10 Ex-Core Neutron Flux Monitoring System

The ex-core neutron flux monitoring system includes neutron detectors located around the core and signal-conditioning equipment located in the control room area.

Ex-core detector channels give the operator source level neutron flux information during extended shutdown periods, initial reactor startup, startups after extended shutdown periods, and following reactor refueling operations. Each channel consists of two sections having multiple BF_3 proportional counters; a preamplifier outside the reactor shield; and a signal-processing drawer containing power supplies, a logarithmic amplifier, and test circuitry. High-voltage power to the proportional counters is terminated when the neutron flux is several decades above the source level to extend the detector life. These channels send information for display and audio count-rate but do not perform control or protective functions. ~~The detectc~~ Ex-core detector channels also send the RRS flux information in the 1-percent to 125-percent power operating range for automatic turbine load-following operations.

7.7.1.11 Boron Dilution Alarm System

The concentration of soluble boronin the RCS affects the control of reactivity in the reactor core. The BDAL receives signals from the ex-core detectors to detect boron dilution events while the reactor is in Modes 3 to 6. The DIAS and DPS monitor the BDAL to ensure detection and alarming of the event.

The alarm setpoint is periodically automatically lowered to a fixed amount above the current neutron flux signal. The alarm setpoint will only follow decreasing or steady flux levels, not an increasing signal. The DIAS and DPS CRTs display the current neutron flux indication and alarm setpoint. A reset capability enables the operator to acknowledge the alarm and initialize the system.

7.7.1.12 Alternate Protection System

APS functions are implemented as part of the process-CCS. The APS augments the RPS to address 10 CFR 50.62 requirements for reducing the risk of anticipated transients without scram (ATWS) and the use of ATWS mitigating systems actuation circuitry (AMSAC).

The APS design includes an alternate reactor trip signal (ARTS) and AFAS that are separate and diverse from the PPS. The ARTS equipment is a diverse means to decrease the possibility of an ATWS, and the AFAS provides added assurance that a source of safety-related feedwater will be available following an ATWS.

The ARTS will initiate a reactor trip when the pressurizer pressure exceeds a nominal setpoint of 16.7 MPa (2420 psia). Turbine-trip signals ~~will~~ can also initiate ARTS if the RPCS is not available. The ARTS circuitry is diverse from that of the RPS. The ARTS design uses a two out of two logic to open the CEDM motor generator output contactors, thus removing motive power to the RTSS, thereby causing the CEAs to drop into the core.

The AFAS will initiate EFW when the level in either steam generator decreases below ~~22~~ 23.4 percent of the 0-1016 cm (0-400 in.) steam generator wide-range level. The EFW components are actuated by sending isolated AFAS signals to the CCL in the ESF-CCS.

The DIAS receives the ARTS and AFAS trip status, pressurizer pressure, and steam generator 1 and 2 level parameters for display. The DPS receives the same data as provided to DIAS.

The APS design requirements in 10 CFR 50.62, "Requirements for Reduction of Risk From ATWS for Light-Water-Cooled Nuclear Power Plants," state that each pressurized-water reactor must have equipment, from the sensor output to the final actuation device, that is diverse from the RTS, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under ATWS conditions. This equipment must perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the RTS. In its original submittal, ABB-CE stated that the EFW actuation circuit design uses the CCL module in the ESF-CCS to satisfy the ATWS rule. ABB-CE did not include sufficient information for the staff to verify that this module is diverse from those in the RTS (DSER Open Item 7.7.1.12-1).

ABB-CE clarified the issue regarding the diversity of the EFW actuation circuitry by stating that the protective system [illegible]tion of the EFWS uses circuitry in the PPS and a component control module in the process-CCS. The process-CCS is diverse and independent from the PPS and the ESF-CCS. Initiation signals generated independently by the two systems are logically

7.7.1.13 Process Component Control System (Process-CCS)

The process-CCS controls non-safety-related pumps, valves, heaters, and fans and other components and sends process variables and CCS status information to the DIAS and DPS for plant monitoring.

The system permits component assignments to independent non-Class 1E subgroup segments to minimize the plant impact caused by component or system-level failures. Standard CCL and I/O links allow for the various types of components to be controlled, as described in Section 7.3. The design includes SCL to supervise subgroups of components and to generate system status information for the DIAS and DPS. The design also includes master transfer capability with isolation to disable all MCR controls and enable component controls for the RSR.

The process-CCS Group Controllers ~~division master processors~~ include a sequencer to automatically start and load the AAC source with essential non-safety loads during LOOP events coincident with a loss of non-safety onsite power. When an EDG is out of service, this sequencer is blocked, permitting the ESF-CCS sequencer to automatically load selected Class 1E division loads.

The process-CCS accommodates both local and remote distribution of I/O multiplexers. The system architecture uses multiple redundant CCL processors with redundant internal data communications.

Both the ESF-CCS (described in Section 7.3) and the process-CCS are microprocessor-based systems with programmable logic for their unique

23-26 °C (73-78 °F) continuous temperature at atmospheric pressure with relative humidity of 20 to 60 percent continuous and 10 Gy (10^3 rad) gamma integrated dose. Environmental Category J encompasses normal and DBA conditions for the control room. The environmental qualification for other control building areas is to be 29 °C (85 °F) continuous temperature at atmospheric pressure with relative humidity of 20 to 100 percent continuous and 10 Gy (10^3 rad) gamma integrated dose.

In the original submittal, ABB-CE had not given the staff sufficient information to evaluate provisions for monitoring the environmental conditions (e.g., temperature, moisture, humidity, chemical pollutants) and conditions that could cause functional degradation (e.g., pipe breaks, fires, loss of ventilation, spurious operation of fire protection systems) within the equipment. Operating experience indicates that such monitors are necessary for alerting plant personnel to equipment operating in an environment beyond its design limits. This was designated DSER Open Item 7.7.1.17-1.

ABB-CE later stated that the HVAC and fire protection systems monitor the environmental space around the equipment cabinets. The HVAC system and the fire protection systems are discussed in Sections 9.4 and 9.5, respectively. DSER Open Item 7.7.1.17-1 ~~will be~~ has been resolved as part of the staff's acceptance of the HVAC design.

The MCPs are compact workstations that integrate miniature backlit component control switches, process controllers, discrete indicators, alarm tiles, message windows, and video display units (CRT, plasma, and electro-luminescent displays) so that both safety-related and non-safety-related display devices are routinely used by the operator.

The MCPs maintain structural integrity so that no control room missile hazards result as a consequence of a seismic event. Any safety-related Class 1E components mounted in the panels are seismically qualified to perform their safety functions.

All NSSS and BOP instruments, controls, and alarms link to the DIAS, DPS, or CCS for routing to the control panels; except for operators' modules dedicated to specific plant components (e.g., PPS, TCS, CEDMCS).

To minimize the possibility of damage to multiple channels within the MCPs or RSPs, the panels employ low-energy circuits (less than 50 V) to the maximum extent possible, fire-retardant materials and smoke detectors, and electrically independent channels of circuits~~, and physical separation or barriers to enhance independence for all circuits with voltages greater than 50 V~~.

If multiple redundant channels are damaged, the MCR circuits are fault isolated from the electronic components with which they interface. All MCP and RSP circuits are passive. Momentary contacts are used for all switches with the memory of control panel commands retained only in the electronic circuits in the I&C equipment rooms. The MCR, RSP, and the I&C equipment rooms are located in separate fire zones.

The DIAS is a segmented, distributed architecture. The system consists of a DIAS-P segment for display of postaccident RG 1.97 variables and DIAS channels N1 through ~~N5~~ N7 for the remaining parameters. Each segment consists of I/O data links and multiplexers, CPUs, and display and alarm devices.

The segmented DIAS includes independent hardware and fault resistance. The DIAS-P is physically independent from the remaining DIAS-N segments and the DPS, so that a single failure will not cause a loss of more than one of the three display methods (DIAS-P, DIAS-N, DPS). The redundant I/O data links and CPUs in each segment allow for transfer to the backup CPU without interrupting the information being displayed on the control panel devices.

Fiber-optic data links provide isolation between the redundant safety-related channel I/O and DIAS CPUs, and between the DIAS CPUs, the MCP I/O multiplexers, and the RSP I/O multiplexers.

The data from both safety-related and non-safety-related sources is scanned at a rate that accommodates the requirements for alarm checking, signal conversion, and signal validation. The DIAS receives signals from the ESF-CCS, the process-CCS, the PCS, the RTSS, ex-core and in-core nuclear instrumentation, the CPCs, the NSSS integrity monitoring system, the PPS, electrical systems components, HJTCs and CETs, and the motor-generator sets.

Input data, calculated values, or parameters for another DIAS segment are available through a data network that interconnects each of the DIAS-N display system. The intelligent display system processes the data for display on the CRTs, and processes operator requests made on the CRT touch screens.

Through host processor, peripheral redundancy, and a distributed design, the DPS accommodates the failure of any single hardware element so that no single failure within the DPS will disable any of the DPS functions. DPS data links acquire plant process data from other plant systems and transmit it to the host processors. The data links between the host processor and plant I+C systems ~~the safety systems~~ are fiber-optic for isolation. ~~All other data links are standard electronic data communication links.~~

Each host processor has a system console for the programmer and consists of a dual CPU. One CPU is dedicated to I/O and demand tasks, the other CPU is dedicated to periodic tasks.

High-speed line printers record information for the programmer, control room operating staff, TSC, and EOF.

The DPS control room operator primarily uses touch screen color CRT workstations and other touch panel devices, such as annunciator tiles. ~~Switches transfer the control of display processor workstations between the primary and backup host processors.~~

All applications are programmed using structured programming rules and techniques. The software comprises modular, structured programs. DPS online operation minimizes reliance on any electromechanical peripherals. All major applications are memory resident and are structured to allow continued execution in the event of a disk, printer, or magnetic tape failure. ABB-CE states that these critical function monitor software modules are verified and validated in accordance with Nuclear Safety Analysis Center 39, "Verification and Validation for Safety Parameter Display Systems." ABB-CE responded to the staff's RAI on software V&V (DSER Open Item 7.7.1.21-1) by submitting a SDP. The staff's acceptance of the SDP, as discussed in Section 7.1.4, resolved DSER Open Item 7.7.1.21-1.

The DPS NSSS application programs give the operator information and alarms to assist in maintaining the plant within specified limits and to evaluate the performance of the reactor core. These applications are:

- core operating limit supervisory system
- CEA position monitoring
- CEA PDIL/PPDIL monitoring
- CEA out-of-sequence monitoring
- CEA deviation monitoring
- CEA trip program
- CEA reassignment
- CEA exposure accumulation
- in-core detector processing
- xenon reactivity prediction
- reactivity balance
- CPC deviation monitoring
- CEAC deviation monitoring
- PPS deviation monitoring
- critical function monitoring

called upon to put on the network to update the active displays. The DPS is designed to provide a 1-second update of dynamic data under worst case loads, based on the maximum number of display devices and the most demanding resultant total for different dynamic parameters to be updated.

The capability to provide for historical data trends on the DPS may call for large data file transfers from one or more of the data acquisition processors. A test will be performed to verify the network response time meets the ~~1-second~~ update criteria for the worst-case file transfer scenarios during maximum data communication loads.

The DIAS and DPS data networks are electrically separated, independent, and diverse from each other. Since the DPS normally provides displays for the complete plant data base, which includes a redundant display of DIAS information, a failure of the DIAS would not increase the communication load of the DPS network.

The staff finds ABB-CE's description and the design of the data networks acceptable. This resolves DSER Open Item 7.7.3-1.

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 and 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

Chapter 9 FSER comments:

p. 9-13	Sect 1, 2nd para: The last sentence should be deleted as the spent fuel handling machine is capable of lifting heavier loads than the fuel assembly and its handling tool.
p. 9-14	Sect 1, 2nd para: The last sentence should be changed as above.
p. 9-30	Sect 2, 1st sentence: Should read "require that the transfer tube valve be closed or that the gate between.....seal."
p. 9-48	1st para, last sentence: Change "crane" to "hoist"
pp. 9-48,49	The 93,5000 gal/hr leakage results from a light load drop (the weight of a fuel assembly falling onto the pool seal from the fuel assembly operating height), not from the heavy load drop. Adverse effects on fuel from the heavy load drop are precluded by the administrative controls mentioned on p. 49, 2nd paragraph, last sentence. Suggest that the last paragraph on p. 48 and first paragraph on p. 49 be revised as follows: "Heavy loads are moved over the pool seal during refueling operations. In order to ensure that water is not lost from the spent fuel pool during a heavy load drop, administrative procedures that will be developed by the COL applicant will require that either the fuel transfer tube valve or the gate between the spent fuel pool and the transfer system canal be closed during movement of heavy loads in the reactor building. The maximum pool seal leakage resulting from a light load drop is 354 m^3/hr (93,500 gal/hr). Should a seal leakage result from a light load drop while moving fuel, the fuel can be moved to a safe location within a maximum of four minutes. Should there be no makeup water to the refueling cavity, it takes approximately four hours to drain the cavity down to the vessel flange. Therefore, there is sufficient water coverage over the fuel assembly during its movement to a safe location. However, to minimize the possibility of having a fuel failure as a result of a heavy load drop, administrative procedures to be developed by the COL applicant will require that no fuel is in the refueling machine when moving heavy loads over the seal."

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.

monitors detect any loss of water shielding over the spent fuel and refueling pools and initiates actuation of the filter mode of the fuel building ventilation system on detection of high radiation.

Based on this design information, the staff concludes that the PCPS includes adequate instrumentation and control features to initiate adequate safety actions should they be needed. This meets the requirements of GDC 63 and is acceptable.

PCPS design provisions for achieving and maintaining adequate water chemistry, sampling of pool water to monitor water chemistry, purification of pool water, maintaining a minimum safe shielding water level, sensing excessive radiation levels and initiating automatic protective actions provide assurance that occupational exposures will be kept as low as is reasonably achievable (ALARA).

The staff identified the following additional Confirmatory Items:

1. In its response to RAI Q410.06b ABB-CE described features of the PCPS design that satisfy Positions C.2.f(2) and [illegible].F(3) of RG 8.8. In the DSER, the staff requested ABB-CE to incorporate this discussion of PCPS design features into CESSAR-DC Section 9.1.3.1. This was Confirmatory Item 9.1.3-7 in the DSER.

   Subsequently, ABB-CE provided Amendment L of CESSAR-DC subsection 9.1.3.1.6 which describes the radiological controls used to minimize area radiation exposure. These include maintenance of a minimum pool water level to ensure radiation exposures are no more than 25 μSv/hr (2.5 mrem/hr) at the pool surface, routing of spent resin transfer lines through shielded pipe chases, design of floor drains to ensure complete collection and routing of radioactive liquid, and the design of the FBVS. <u>In addition, all valves in contact with spent fuel pool water are made of austenitic stainless steel or an equivalent corrosion-resistant material. All piping in contact with spent fuel pool water is austenitic stainless steel</u>. Based on the incorporation of this information into the CESSAR-DC, Confirmatory Item 9.1.3-7 is resolved.

These items are discussed in Cessar-DC Sections 9.1.3.2.2.8 (VALVES) and 9.1.3.2.2.9 (Piping)

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, and Figure 9.4-2, respectively.

3.2-1

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, 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.

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

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 the environment to ensure that all potentially radioactive releases are monitored before discharge. The system is a once-through cycle type.

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

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.

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 generators (EDGs) 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.

Non-Credited

The divisional exhaust filtration unit consists of a moisture eliminator, prefilter, electric preheater, ~~non~~ 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.

Redundant components of the safety-related equipment room cooling systems are physically separated and protected from internally generated missiles. A pipe break in the same safety-related mechanical train is the only possible means of affecting the safety-related CWS. Therefore, when subjected to pipe break effects, the components are not required to operate because the served equipment is located in the same space as the cooling components. All safety-related equipment is rated seismic Category I and located in a seismic Category I building. The components are protected from tornadoes, and intake and exhaust vents are protected from rain, snow, and ice. Failure of the non-safety-related supply units will not affect the safety function of the safety-related units.

The ESF exhaust filtration trains minimum instrumentation are listed in CESSAR-DC Table 9.4-3A. Air flow rates of fans, operating status of fans, temperature and flow rate of chilled water, damper positions/alignment, 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. The equipment is inspected periodically and the design allows for in-service inspection.

Because safety-related components are classified as seismic Category I and located within a seismic Category I structure, the ventilation system meets the requirements of GDC 2.

By virtue of design with respect to maintenance of environmental conditions and consideration of dynamics effects, the system meets the requirements of GDC 4.

As indicated in above, the system is consistent with the requirements of RGs 1.140 and 1.52 for exhaust system cleanup and filtration and, therefore, meets the requirements of GDC 60.

ABB-CE committed to incorporate in the CESSAR-DC its response to staff RAI Q410.119 concerning the in-service testing, exhaust fan failure, fresh air intakes, and RGs 1.140 and 1.52 conformance. This was identified as Confirmatory Item 9.4.5-1 in the DSER. Subsequently, ABB-CE stated that: (1) all safety-related subsphere ventilation system components are designed to permit in-service 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).

adsorber as shown in CESSAR-DC Figures 9.4-1 and 9.4-6. The location and sequence of equipment of the high-purge subsystem parallels that of the above low-purge system with the following exceptions: (1) the high volume purge system has two penetrations for each high purge supply and exhaust train; and (2) each penetration has two failed-closed pneumatically operated isolation dampers, one in the annulus and one in containment.

In response to staff Q410.120, ABB-CE stated that the containment high volume purge system is not an engineered safety feature system. During a postulated fuel handling accident, the charcoal filtration is credited with filtration of the release, but no credit is allowed for release reduction resulting from containment isolation or mixing in the containment atmosphere proceeding the release. ABB-CE committed to incorporate the response in the SSAR. This was identified as Confirmatory Item 9.4.6-1 in the DSER. Subsequently, ABB-CE stated in CESSAR-DC Sections 9.4.6.1 and 9.4.6.3 that the dose analysis demonstrating conformance with 10 CFR Part 100 limits following a fuel-handling accident and a control element ejection accident, only takes credit for the HEPA filtration. No credit is taken for the ~~charcoal~~ carbon adsorbers in the either of the containment exhaust paths.

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 CC&VS.

ABB-CE revised CESSAR-DC Table 3.2-1 in response to staff RAI Q210.1 to show that the safety-related components (eleven containment isolation dampers and the associated penetration ductwork from the containment penetration to the filter trains) are Safety Class 2 and seismic Category I, and that the quality assurance requirements of 10 CFR Part 50, Appendix B are applicable. The remaining components are not safety related. All eleven containment isolation dampers are normally closed, are designed to fail closed, and receive a containment isolation signal to close. Additionally, the high volume purge system remains sealed closed during power operation. The containment purge exhaust system is isolated on high radiation signal or high relative humidity signals.

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 in-service 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.

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 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.

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 "Air Conditioning and Ventilation Systems" 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.

-90-16

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.

9.5.4.1.2

In a letter of February 28, 1992, EPRI advised ABB-CE to add Section ~~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

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)**

INSERT E:

"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 1, 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. ①

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.~~ The minimum available NPSH is the smaller of either 25 percent ~~of~~, or 3m (10 ft) greater than, the required NPSH specified by the pump vendor. 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.

ADD INSERT A

(2)

ADD INSERT B

(3)

greater than

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, instrumentation 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 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 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 condenser and aftercooler~~; boric acid concentrator ~~distillate cooler, condenser, and condenser vent cooler~~; ~~non-essential~~ normal 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 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

ADD INSERT D

4

5

6

7

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 interface. These valves automatically close on a low-low surge tank level, preventing 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 divisions, 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 instrumentation 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 ~~upper bearing~~ motor 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 interface. These valves automatically close on a low-low surge tank level, preventing 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 divisions, 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 instrumentation 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.

(8)

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. 8

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

8

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.

7

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. 9

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

chiller, a heat exchanger,

Each 100-percent-capacity ECWS division consists of a ~~chilled water refrigeration unit~~, ~~a circulating~~ two chilled water pump, & an expansion ~~a compression~~ tank, ~~control valves, instrumentation, and piping~~. ~~Additionally, there is a~~ The ECWS heat exchanger and associated chilled water ~~heat exchanger~~ 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 for the pump that serves the essential chiller ~~ECWS pump~~. 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 Alternate ~~standby 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 ~~T~~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 consist of ~~comprises~~ 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

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.~~ 11

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. 12

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 1E 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 6

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

3

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 design 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 Powered ~~power~~ 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

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.

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.

abstract and verified that the applicant addressed Position C.2 by providing test method 3.5 which requires testing of the RHR system isolation alarms and signals. The staff finds this to be acceptable.

b. RHR System pressure relief (Positions C.3)

ABB-CE should modify CESSAR-DC Section 14.2.12.1.21, test method 3.7 to state "verify the <u>relieving capacity and</u> setpoint of the Low Temperature Over-Pressure Protection (LTOP) relief valves" to satisfy the guidance RG 1.139, Position C.3. The applicant responded by stating that the LTOP relief valves relieving capacity is conducted during bench testing before the valves are installed in the RHR System. Since the bench test verifies the relieving capacity, preoperational testing of the LTOP relief valve relieving capacity is not necessary. The staff stated that to conform to RG 1.139, the applicant should add this information to the prerequisite section to state that "the LTOP relief valves relieving capacity is verified by bench testing." The applicant will incorporate this information into the test abstract prerequisite section into the CESSAR-DC. The staff finds this response acceptable pending incorporation into the CESSAR-DC. This is identified as part of FSER Confirmatory Item 1.1-1 (see Chapter 1 of this report).

To be included in Amend V

14.2.13 Security Considerations

The startup test program described in CESSAR-DC Section 14.2 includes security lighting system and security radio system tests. (CESSAR-DC Section 9.5.2 provides a security radio system for offsite communications.) Security "lock-down" of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) were not addressed. In its December 17, 1991 response to RAI Q500.31, the applicant stated, "Security 'lockdown' of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) is considered sensitive information which may be withheld from the public by the directive of 10 CFR 2.790(d). Full disclosure and description of these sensitive systems and their prerequisite testing will, however, be a part of the site security plan to be submitted by the utility." The staff finds it acceptable that detailed description and NRC review of the rest of the security system and its test and acceptance criteria be deferred to the COL applicant's security plan submittal required by 10 CFR 50.34. This was identified as COL Action Item 14.2.13-1. By letter dated January 26, 1993, ABB-CE identified in CESSAR-DC Sections $\boxed{14.2.13}$ that the COL applicant will provide site-specific security, contingency, and guard training plans. The inclusion of this information into the CESSAR-DC resolves this issue.

There is no Section 14.2.13 We Added it to 14.2.7.5

In RAI Q500.31, the staff noted that the security lighting system test described in CESSAR-DC Section 14.2.12.1.85 was incomplete in that it does not address testing on loss of normal power nor testing of its adequacy for support of closed-circuit television (CCTV) security functions. In its December 17, 1991 response, the applicant committed to amend subsections of the security lighting system test method to read:

3.3 Demonstrate that loss of normal power results in proper activation of the Security Lighting System for each affected room.

3.4 Demonstrate the Security Lighting System provides adequate illumination levels, including, but not limited to, those required to support plant Closed Circuit TV security functions.

The staff found that these proposed changes did not adequately address its concerns regarding demonstrating that the system as installed has the capability described in the CESSAR-DC. CESSAR-DC Section 9.5.3.1 states that the security lighting system will provide "illumination required to monitor isolation zones and all outdoor areas within the plant protected perimeter, under normal conditions as well as upon loss of all ac power." The proposed change addressed demonstrating lighting adequacy on loss of power "for each affected room" but not where "required to monitor isolation zones and all outdoor areas within the plant protected perimeter." This was identified as DSER Open Item 14.2.13-1. By letter dated April 21, 1993, ABB-CE modified CESSAR-DC Section 14.2.12.1.85 to require demonstration of the adequacy of illumination for CCTV security functions on loss of normal lighting power for monitored isolation zones and outdoor areas within the plant protected out-of-scope portions of the systems of the design are contained in CDM Section 4.0. ABB-CE provided an entry in Section 2.0 ~~this section~~ for every system of the design to define the full scope of the design. Confusing as written. Implies Section 4.0 has these entries.

The detailed design information for the System 80+ is contained in the CESSAR-DC. The staff's safety evaluations for the design are based on the System 80+ design material in the CESSAR-DC, and are provided in the appropriate sections of this report.

The information in this section of the CDM is derived from the detailed design information contained in the CESSAR-DC. The purpose of the ITAAC, which are part of the CDM, is to verify that a plant that references the design certification is built and will operate in accordance with the design certification. Consequently, there is no design information presented in the CDM system design descriptions and related ITAAC, or CESSAR-DC Section 14.3 that is not also contained in the various sections of the CESSAR-DC. The staff did not base its safety decisions on the information in the CDM and, therefore, this section of the report contains no safety evaluations for the design.

14.3.2.1 Design Descriptions

The design descriptions address the most safety-significant aspects of each of the systems of the design, and were derived from the detailed design information contained in the CESSAR-DC. The design descriptions include the figures associated with the systems. ABB-CE's selection criteria and methodology for the system design descriptions are specified in CESSAR-DC Section 14.3.2.1. In its review of the material, the staff followed the general guidance for the reviews specified in the SRM related to SECY-90-377, as discussed previously in the introduction to Section 14.3 of this report.

The Tier 1 design descriptions will serve as a facility lifetime commitment. Once completion of ITAAC and the supporting design information demonstrate that the facility has been properly constructed, it then becomes the function of existing programs such as the technical specifications, the in-service inspection and in-service testing program, the quality assurance program, and the reliability assurance program, to demonstrate that the facility continues

8. Interlocks
9. Class 1E electrical power sources/divisions
10. Equipment to be qualified for harsh environments
11. Interface requirements
12. Numeric performance values
13. Accuracy and quality of figures

Additionally, standard ITAAC entries were utilized to verify selected issues, where appropriate. Examples of these included basic configuration, physical separation, and divisional power supplies. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC was reviewed to ensure appropriate treatment in the CDM.

Environmental qualification (EQ) of safe-shutdown equipment is verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC is discussed in the General Provisions section of the CDM. Verification includes type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.

The task group reviewed integrated plant safety analyses such as fires, floods and missile protection to ensure they were adequately addressed in the CDM. The insights from these analyses that were addressed in the CDM are contained in CESSAR-DC Section 14.3. The issues of floods, fires, missiles, pipe failures, and environmental protection are verified by the ITAAC on a system-specific basis, rather than generically. Divisional separation (both physical and electrical) is the primary means of ensuring protection of safety-related equipment from these events. Verification of divisional separation is performed as part of both individual system ITAACs and building ITAACs. Physical and electrical separation is verified in each safety-related system ITAAC and divisional barriers are verified in the reactor and control building ITAACs.

Nuclear Island Structures

for treatment in the CDM. The supporting information regarding the detailed design and analyses remained in the CESSAR-DC. For many of the design features, it was impractical to test their functionality. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, was considered sufficient CDM treatment.

The staff determined that the detailed supporting information in the CESSAR-DC for the Nuclear Fuel System in the CDM, if considered for a change by a COL applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question. This supporting information includes the fuel design, the control element assembly (CEA) design and the initial core design. Thus, the staff has concluded that any changes to the initial reference design of these areas from that presented and evaluated in CESSAR-DC Chapter 4 will require prior NRC review and approval, with the exception of the design features and design parameters in Tables 4.2-1 and 4.2-2 and changes covered by applicable NRC-approved topical reports. Furthermore, prior NRC review and approval will be required to change the NRC-approved analysis methods used to demonstrate conformance of the fuel design, the CEA design and the initial core design to the design limits given in CESSAR-DC, with the exception of those changes to analysis methods that are covered by applicable NRC-approved topical reports. The specific fuel, CEA, and core designs presented in CESSAR-DC Chapter 4 will constitute, based on staff review and approval, an approved design that may be used for the COL first cycle core loading, without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review that specific fuel, control rod, and core design analyses as described in CESSAR-DC Chapters 6 and 15.

IS THIS OK

Examples of the issues that the task group examined for treatment in Tier 1 included net positive suction head for key pumps (specified in the applicable systems), and intersystem LOCA (the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary is specified in the design descriptions of the applicable systems). The task group also reviewed the ITAAC for consistency with the initial test program described in CESSAR-DC Chapter 14.

Chapter 7 of the CESSAR-DC. The individual systems that contained the sensors for the displays, controls, and alarms were reviewed to ensure that standard ITAAC entries were used to verify their function. The design processes and acceptance criteria for I&C equipment contained in CDM Section 2.5, particularly the verification and validation aspects of the I&C acceptance criteria, will verify proper operation of the I&C aspects of the equipment. Similarly, the design processes requirements for HFE contained in CDM Section 2.12, particularly the availability and suitability verification, and validation aspects of the control room and remote shutdown room, will verify proper design of the equipment for human factors considerations.

The staff conducted a complete and thorough review of the System 80+ CDM material to ensure that the general criteria of the eight program elements in the HFE PRM were appropriately addressed in the Tier 1 CDM. The Tier 2 CESSAR-DC material contains more detailed guidelines and applicable guidance documents. The staff also conducted a review of the CESSAR-DC material to ensure that the specific acceptance criteria in the HFE PRM were appropriately addressed. The staff reviewed the CDM and CESSAR-DC Section 14.3.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to CDM development provided by the Commission.

The material in CESSAR-DC Chapter 18 provides design information and defines design processes for the four remaining HFE PRM elements that are acceptable for use in meeting the acceptance criteria in the CDM. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for the control room is based on the design processes and acceptance criteria material as described in CDM 2.12 and Chapter 18 of the CESSAR-DC. Consequently, the staff indicated in Chapter 18 and Section 1.11 of this report that any proposed changes to CESSAR-DC Sections 18.5, 18.6, 18.7, 18.8, and 18.9 constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation. These sections contain: Tier 2*

- Section 18.5, "Functional Task Analysis"
- Section 18.6, "Control Room Configuration"

Section 3.10.1 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the dynamic qualification of equipment. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.10 of this report. A listing of this information is also contained in Section 1.11 of this report. Tier 2*

Valves - The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. Specifically, for MOVs, a special inspection is required as a part of the basic configuration check to verify the records of vendor tests that demonstrate the ability of MOVs to function under design conditions. In addition, in-situ tests are required for MOVs and check valves in each system ITAAC. These tests will be performed during the initial test program. The material in CESSAR-DC Section 3.9.6.2.2 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design basis capability. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.9.6 of this report. A listing of this information is also contained in Section 1.11 of this report. Tier 2*

Piping - The verification of the overall piping design including the effects of high-energy line breaks and the application of leak-before-break (as applicable) is performed in conjunction with the piping DAC. The as-built piping system is required to be reconciled with the design commitments. The material in CESSAR-DC Section 3.12 provides detailed supporting information for the CDM regarding the analysis methods and design criteria to be used by the COL applicant or licensee to complete the piping design. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation Tier 2* of the change. This material supporting the CDM is discussed further in Section 3.12 of this report. A listing of this information is also contained in Section 1.11 of this report.

<u>Review of the System 80+ Structural Design Integrity</u>

Nothing in Tier 1 on service bldg.

IMPLIES A SEPARATE BUILDING

The scope of structural design covers the major structural systems in the System 80+ plant including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, and radwaste building). The RPV, piping systems, and primary containment are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety.

In establishing the top level requirements for structural design, the staff used the General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for the Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 16, "Containment Design," and GDC 50, "Containment Design Basis."

> GDC 1 requires, in part, the need for structures, systems and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

> GDC 2 requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

structural design basis. A summary of the top level structural design requirements for the major structural systems that are verified by the structures and systems in CDM Section 2.0 and the piping design information in CDM Section 3.3 is provided below.

For ~~the~~ System 80 this can be either 3.0 or 3.1 not 3.3. (3.3 is the ABWR)

Pressure Boundary Integrity

To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAAC were established to verify the pressure boundary integrity of the RPV, piping, and primary containment for the System 80+. GDC 16 and 50 apply to the primary containment and GDC 14 applies to the RPV and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle.

For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III are required by ITAAC. For the primary containment, a structural integrity test is required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Because the requirements of GDC 14, 16, and 50 do not apply to the reactor, control, turbine, service, and radwaste buildings, ITAAC were not required to verify the pressure integrity for these other buildings.

Normal Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena.

For piping systems, ITAAC require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). For the RPV, the fabrication is performed primarily in the vendor's shop where adherence to design drawings is tightly controlled. Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV were deemed to be inappropriate. Instead, ITAAC verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin.

Similarly, for safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results are to be documented in a structural analysis report, the scope and contents of which are described in the CESSAR-DC. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings. The verification of the service and turbine building design for normal loads is subject to the requirements of the normal 10 CFR Part 50, Appendix B quality assurance program.

Seismic Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads is addressed by ITAAC established for verifying the basic configuration of systems.

As discussed above for normal loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). For the RPV, ITAAC for the as-built reconciliation of seismic loads for the RPV were deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code-required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings. However, because the leakage path for fission products includes components within the turbine building, the turbine building is required to withstand the effects of a safe-shutdown earthquake. Therefore, ITAAC were established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any structures or equipment located adjacent to or within the turbine building.

For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform their safety-related functions was assessed. Because the design detail and as-built and as-procured information for many non-safety related systems (e.g., field-run piping and balance-of-plant systems) are not required for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for assuring acceptable II/I interactions and a commitment for the COL applicant to describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions have been specified in the CESSAR-DC.

Suppression Pool Hydrodynamic Loads

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to suppression pool hydrodynamic loadings which include those loadings associated with safety relief valve discharge and loss-of-coolant accident (LOCA) loadings. Component qualification for suppression pool hydrodynamic loads is addressed by ITAAC established for verifying the basic configuration of systems.

I don't know where this is covered in ITAAC.

As discussed above for seismic loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design-basis loads (which include suppression pool hydrodynamic loads). For the RPV, ITAAC verify that the ASME Code-required reports exist to ensure that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For the reactor building and primary containment including the internal structures, ITAAC require an analysis for reconciling the building as-built configuration with the structural design basis loads (which include suppression pool hydrodynamic loads). The as-built analysis results are to be documented in a structural analysis report as discussed above. The effects of suppression pool hydrodynamic loads do not extend beyond the reactor building, and, thus, ITAAC are not required to verify these loadings for the other System 80+ building structures.

Looks like ABWR wording

ITAAC also require the verification of the horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design.

Flood, Wind, Tornado, Rain, and Snow

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow.

These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings. The verification of the service and turbine building design for these loadings is subject to the requirements of the normal 10 CFR Part 50, Appendix B quality assurance program.

For flooding, ITAAC also require inspections to verify that water-tight doors exist, penetrations (except for water-tight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC require that external walls below flood level are equal to or greater than 0.6 m to protect against water seepage and penetrations in the external walls below flood level are provided with flood protection features.
(These are not in our ITAAC at present!)

Pipe Break

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks is addressed by ITAAC established for verifying the basic configuration of systems.

? Don't recognize this. Looks like ABWR

For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses.

In addition. ITAAC have been established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings either because they are not safety-related structures or there are no high-energy lines located within the structure.

Codes and Standards

To ensure that the applicable requirements of GDC 1 have been adequately addressed, ITAAC were established to verify that appropriate codes and standards were used in the design and construction of safety-related systems and components. In general, the staff established only those codes and standards endorsed by the regulations under 10 CFR 50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components was established as the code for the design and construction of System 80+ piping systems and the RPV. For safety-related building designs, the staff based its safety findings on audits of System 80+ design calculations which relied on specific codes and standards. These codes and standards are contained in CESSAR-DC Section 3.8, and were identified in Section 3.8 of this report as material that, if considered for a change by an applicant or licensee that references the certified System 80+ design, would

Tier 2*

constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. These are also listed in Section 1.11 of this report.

Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code.

No ITAAC were established for System 80+ building structures because the codes and standards used for their design and construction are not included in 10 CFR 50.55a.

<u>As-built Reconciliation</u>

To ensure that the final as-built plant structures are built in accordance with the certified design as required by 10 CFR Part 52, structural analyses will be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses will be documented in structural analysis reports. Structural analysis reports will be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and turbine buildings. The detailed supporting information on what is required for an acceptable analysis report is contained in CESSAR-DC Chapter 3.

Similarly for piping systems, an as-built analysis will be performed using the as-designed and as-built information. ITAAC will verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed.

The dimension are for INFO ONLY

For the RPV, the key dimensions of the RPV system will be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions are provided in the certified design description.

For component qualification, tests, analyses, or a combination of tests and analyses will be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These tests and analyses will be performed as a part of ITAAC to verify the basic configuration of the system in which the equipment is located.

Conclusions

On the basis of the above discussion, the staff concludes that the structural design integrity of the SSCs important to safety in the System 80+ standard plant is adequately verified by ITAAC by ensuring that top-level design, fabrication, testing, and performance requirements for SSCs important to safety are satisfied. As discussed above, these top-level requirements constitute the bases for concluding that the ITAAC are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria are met, a facility that references the design has been constructed and will operate in conformance with the design certification and applicable regulations for assuring the integrity of the System 80+ structural design.

provided for the primary automatic logic. Diversity is provided in the form of hardwired backup for reactor trip, diverse display of important process parameters, defense-in-depth arrangement of equipment, and other equipment diversity.

The staff conducted a complete and thorough review of the System 80+ CDM material to ensure that the SRP guidelines and Commission guidance for I&C design were appropriately addressed in both the CDM and the CESSAR-DC. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the System 80+ I&C design. The CESSAR-DC material contains more detailed guidelines and applicable documents. The staff reviewed the CDM and CESSAR-DC Section 14.3.2 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to CDM development provided by the Commission.

Selected material in CESSAR-DC Chapter 7 provides design information and defines design processes that are acceptable for use in meeting the I&C acceptable criteria in the CDM. This material includes design information regarding hardware and software changes, commercial dedication, equipment qualification, and electromagnetic compatibility. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for I&C systems is based on the design processes and acceptance criteria material in the CDM and the CESSAR-DC. Consequently, the staff indicated in Section 7 of this report that any proposed changes to the appropriate CESSAR-DC sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation. These issues are also listed in Section 1.11 of this report. Tier 2*

The certified design description and design development process continue for the lifetime of the plant. Any safety-related software that is changed or added after plant startup is required to either be developed using the certified design development process described in the CDM, or the licensee must submit a design process (together with the design bases) description that

CESSAR-DC Section 14.3.3.2 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to CDM development provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the background of Section 14.3 of this report.

Selected material in CESSAR Chapter 3 and Appendix 3.9A provides design information and defines design processes that are acceptable for use in meeting the piping DAC in the CDM. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for piping systems is based on the design processes and acceptance criteria material in the DAC and the CESSAR-DC. Consequently, the staff indicated in Section 3.12 of this report that any proposed changes to the appropriate CESSAR-DC sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation. This information is also listed in Section 1.11 of this report. Tier 2*

14.3.3.1.2 Approval of the Piping Design DAC

Based on the staff's review of the material in the CDM, and a review of the selection methodology and criteria for the development of the CDM contained in CESSAR-DC Section 14.3.3.2, the staff concludes that the material in the CDM is necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria met, a facility referencing the certified design will be constructed and will operate in conformity with the design certification, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

14.3.3.2 Radiation Protection DAC

14.3.3.2.1 Review of the Radiation Protection DAC

The radiation protection aspects of the System 80+ design are provided in CESSAR-DC Chapter 12, "Radiation Protection," and together with the associated DAC in CDM Section 3.2, "Radiation Protection," are evaluated in Chapter 12 of

The analysis methods used for transient and accident analyses are normally reviewed on a generic basis. ABB-CE topical reports describing analytical methods and the associated NRC approval letters (as stated in the response to the staff request for additional information (RAI) Q440.89) were incorporated into CESSAR-DC Sections 6.3.3 and 15.0.3 and are listed in Table 15.1-3 of this report. The approved methods for non-LOCA analysis include the following computer codes:

- The CESEC-III (Refs. 1 and 2) code: Calculates system parameters including core power, flow, pressure, temperature and valve actions during a transient.

- The TORC (Ref. 3) and CETOP (Ref. 4) codes: TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative channel axial flow that is used to calibrate CETOP. TORC or CETOP ~~is~~ are used for all DNBR calculations using the CE-1 critical heat flux correlation.

- The HERMITE (Ref. 5) code: HERMITE is used to determine short-term response of the reactor core during the postulated total Loss of flow event and reactor coolant pump rotor seizure event.

- The COAST (Refs. 6 and 7) code: Calculates the time-dependent reactor coolant mass flow rate in each loop during RCP coastdown transients.

- The STRIKIN-II (Ref. 8) code: Calculates the cladding and fuel temperatures for an average or hot fuel rod.

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

- The CEFLASH-4A (Ref. 9) and CEFLASH-4AS (Ref. 10) codes: CEFLASH-4A and CEFLASH-4AS determine the primary system hydraulic parameters during blowdown phase for the analysis of a large-break LOCA and a small-break LOCA, respectively.

When we can change the fuel design and under what conditions is covered by the Tier 2* we agreed upon!

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. [redacted]. 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 Tier 2 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 but rather took it as an additional single failure to show limiting fault criteria were not exceeded. However, GDC 17 in 10 CFR Part 50, Appendix A requires, in part:

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

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

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

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

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

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

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

4. The most positive MTC at full power is changed from 0.0 to -1.8×10^{-4} delta-rho/°C (0.0 to -0.1×10^{-4} delta-rho/°F). At zero power, the MTC is reduced from 0.9×10^{-4} delta-rho/°C ($+.5 \times 10^{-4}$ delta-rho/°F) to 0.0.

5. The CEDM coil delay time is reduced from 0.8 seconds to 0.5 seconds.

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

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

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

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

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

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

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

4. Item 4 -- The values of the MTC is within the limits of TS 3.1.4.

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

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

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

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

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

Table 15.1-3. Computer Codes Used in the Safety Analyses

Code	Documentation	NRC approval letter
Non-LOCA analysis		
CESEC-III (system code)	LD-82-001	Ref. 15
CETOP-D (Rev. 1)	CENPD-161, CEN-160-S-P	Ref. 16
TORC (thermal-hydraulic code)	CENPD-206	Ref. 17
HERMITE (neutronic code)	CENPD-188	Ref. 18
COAST (RCS pump flow cooldown code)	CENPD-98	Ref. 19
STRIKIN-II (fuel behavior code)	CENPD-135, Supplements 2 and 4	Ref. 20
Small-break LOCA analysis		
CEFLASH-4AS (systems blowdown code)	CENDF-133, Supplements 1 and 4	Refs. 21 and 22
COMPERC-II (reflood system code)	CENPD-134, Supplements 1 and 2	Refs. 21 and 23
STRIKIN-II (fuel behavior code)	CENPD-135, Supplements 2, 4, and 5	Refs. 21, 24, and 25
FATES3 (fuel gap conductivity code)	CFNPD-139, CEN-161(B)	Refs. 26 and 27
PARCH Steam cooling (~~pool boil~~ing heat Transfer Code)	CENPD-138, Supplements 1 and 2	Refs. 21 and 28
Large-break LOCA analysis		
CEFLASH-4A (system blowdown code)	CENPD-133, Supplements 2, 4 and 5	Refs. 21 and 23
COMPERC-II	CENPD-134, Supplements 1 and 2	Refs. 21 and 23
STRIKIN-II	CENPD-135, Supplements 2, 4 and 5	Refs. 21, 24, and 25

15.2 Transient Analyses

For the System 80+ design, ABB-CE analyzed all events described in SRP Chapter 15 and presented the limiting event or event combination for each category in analytical detail in CESSAR-DC Chapter 15. For non-limiting events, ABB-CE provided qualitative discussions explaining why the events are not limiting. The staff's evaluation of the system responses and thermal-hydraulic behaviors of the analyzed transients is discussed in Section 15.2 for transient analyses and in Section 15.3 for accident analyses. The staff's evaluation of the radiological consequences for various postulated events is presented in Section 15.4 of this report.

15.2.1 Increase in Heat Removal by the Secondary System

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

- decrease in feedwater temperature (SRP Section 15.1.1)
- increase in feedwater flow (SRP Section 15.1.2)
- increase in steam flow (SRP Section 15.1.3)
- inadvertent opening of a steam generator (SG) relief or safety valve (SRP Section 15.1.4)

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

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressure.

- Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above 95/95 DNBR safety limit.

- A transient of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

- A transient of moderate frequency in combination with single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding.

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

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

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

15.2.2 Decrease in Heat Removal by the Secondary System

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

- loss of external load
- turbine trip
- loss of condenser vacuum
- main steam isolation valve closure

* loss of non-emergency ac power to the station auxiliaries
* loss of normal feedwater flow

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

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

In compliance with GDC 17 that requires a loss of offsite power (LOOP) to be considered in the analysis, ABB-CE assumed that the LOOP occurred coincidently with the initiation of the LOCV event, which results in a simultaneous turbine trip. The LOOP causes the four RCS pumps to coastdown, which in turn, results in a reactor trip signal generated by the low pump shaft speed. This reactor trip signal occurs earlier than that generated by the high pressurizer pressure signal from the LOCV event without LOOP. Consequently, ABB-CE indicated that with respect to peak pressure, the LOCV event with power available has a longer reactor trip delay time for the RCS pressure to increase and ,thus, is more limiting than the event with LOOP. With respect to fuel performance, the LOCV event with ~~LOOP, similar to the loss of~~ offsite power available

~~non-emergency power to the station auxiliaries,~~ results in a lowest minimum DNBR. ABB-CE evaluated the single failures listed in CESSAR-DC Table 15.0-4 and concluded that no single failure will result in a lower minimum DNBR or a higher peak RCS pressure than that for the LOCV event with ~~combination of a LOOP following a turbine trip.~~ offsite power available.

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

15.2.3 Decrease in Reactor Coolant Flow Rate

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

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

A loss of power to all reactor coolant pumps produces a reduction of coolant flow through the reactor core. The reduction in coolant flow rate causes an increase in the average coolant temperature in the core and a decrease in margin to DNB. A low ~~DNBR~~ RCP shaft speed trip is initiated by the core protection calculator (CPC) a 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 a loss of 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.

nuclear annex; and (3) the pressurizer low-level, nuclear annex sump high level and the volume control tank low level alarms.

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

minimum

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

enthalpy & flashing

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

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

staff includes the evaluation for the post-LOCA long term cooling in Section 15.3.8 of this report.

Boron dilution during SBLOCAs

Existing experimental evidence and recent analysis results show that an inherent mechanism for boron dilution in the PWR RCP loop seals could exist for events (including small break loss of coolant accidents, SBLOCAs) that involve heat removal by reflux cooling. The deborated water in the RCP loop seals could be transported to the core through natural circulation processes or startup of RCPs. Injection of the deborated water into the core would be a significant reactivity addition that could possibly result in damage to the core. ABB-CE was requested to address the applicability of this boron dilution event to the System 80+ design and provide resolutions to this issue.

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

A bounding analysis was performed without crediting any of the mixing of borated and unborated water which is expected to occur in the RCS. Instead, the condensate was assumed to enter the core as ~~a~~ an unlimited size slug of pure water, with the ~~effective size of four cold leg loop seals [each approximately 5.66 m^3 (200 ft^3) in volume]~~ moving at a natural circulation flow rate consistent with that of a small break at the time of RCS refill using ECCS injection.

The mixing calculations suggested a significant degree of mixing would occur.

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

Subsequently, ABB-CE presented additional information to the staff regarding this issue at a meeting held on February 10, 1994. Information was provided on the use of the FLUENT code for mixing calculations and on the actual flow skirt and lower internals configuration under the RCS conditions cited above. Specific conservatism regarding such things as jet stream effects on mixing were clarified. The effect of time in cycle was discussed and showed that only during 1/3 of the fuel cycle was a return to critical possible even with a pure slug. Also, the volume of condensate formed was related to the number of SI pumps available. Therefore, cases of larger condensate production were found to be of lower probability since it is more probable that all SI pumps would be available. In addition, ABB-CE has further modified the System 80+ EOGs to require technical support center approval prior to RCP restart and has re-ordered the EOG steps to emphasize verification of natural circulation prior to consideration of RCP restart. Finally, ABB-CE indicated that should a prompt critical excursion which breaches the vessel occur, the containment is expected to remain intact because of the availability of containment sprays and cavity flooding.

The staff considers this issue to be technically resolved. It will remain a confirmatory item, however, pending formal documentation of the information presented at the meeting, incorporation into the CESSAR-DC, and update of the System 80+ EOGs. This issue is identified as part of FSER Confirmatory Item 1.1-1.

The SBLOCA analyses will be covered in Amend +. SER needs to advise Fin & T. RPR how to handle. Eric!

15.3.8 Post-LOCA Long-Term Cooling

Long-term cooling (LTC) initiates when the core is quenched after a LOCA and terminates when the plant is secured. The objectives of LTC are to maintain the core at safe temperature level and to avoid the precipitation of boric acid in the core region. In CESSAR-DC Section 6.3.3.4, ABB-CE described the LTC methods (Ref. 29) for the System 80+ design.

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

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

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

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

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

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

indicated

Since previously approved methods were used to demonstrate an adequate margin available for the post-LOCA LTC, this satisfies the long-term core cooling acceptance criteria of 10 CFR 50.46. However, the original analysis credited the auxiliary pressurizer spray, which is a non-safety-grade system, for the RCS cooldown and assumed the SCS entry conditions to be 4.19 x 10^3 kPa and 204 °C (608 psia and 400 °F). The entry conditions for the System 80+ SCS are 2.76 x 10^3 kPa and 177 °C (~~400~~ psia and 350 °F). ABB-CE was required to reanalyze the post-LOCA LTC using only safety-grade systems and using the SCS for LTC based on the design initiation temperature and pressure. This was designated as part of DSER Open Item 15.3.1-2.

450

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.

0.056m² (0.6ft²)

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 is ~~0.065m² (0.7 ft²)~~. ABB-CE determined that the minimum DNBR experienced throughout the event is less than 1.24 and that less than 0.22 ~~0.15~~ percent fuel failure would result. DNBR is minimized at a break size of 0.02 m² (0.2 ft²). A total of ~~57,700~~ kg (~~127,000~~ 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. During

Secondary 65,888 145,023

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. In ABB-CE's analysis, thyroid doses at the exclusion area boundary of ~~0.25~~ 0.195 Sv (~~25~~ 19.5 rem) were computed. ABB-CE also computed a whole body dose of ~~7.6~~ 3.3 x 10^{-4} Sv (~~0.076~~ 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 1.0×10^{-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.

main steam safety valves and

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 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 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 are provided inside containment, which limit the leakage of reactor coolant outside containment to a value such that regulatory acceptance criteria from SRP 15.6.1 are satisfied. On this basis, DSER Open Item 15.4.2.4-1 is resolved.

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

more flashing of the

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

which maximizes minimized

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

15.4.2.5.2 Steam Generator Tube Rupture

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

2.55

Using an alternative process, ABB-CE concluded that the maximum allowable dilution factor is ~~1.48~~ x 10^{-6}. 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 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 a~~r~~gumentation 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

[redacted]

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

Subject: Review of Chapter 17 of System 80+ FSER

I have reviewed Chapter 17 of FSER and found no major errors. John Pasquenza also reviewed Chapter 17.1 (QA) and found no errors. [redacted] The following specific changes are suggested because the current wording could be interpreted incorrectly or is misleading.

Page 17-2, Section 17.1.3, 1st paragraph, last 2 sentences:
These sentences imply that ABB-CE accidently left certain RGs in Table 1.8-1. The purpose of Table 1.8-1 was to address all RGs, including those that were superseded by RG 1.28. The last sentence is very negative and could be changed. Suggested wording for these two sentences are:

"A review of the QA-related RGs are listed in Table 1.8-1. All DSER issues have been closed and a more detailed discussion follows:"

Page 17-5, 2nd line from top of page, last sentence:
This sentence is negative and suggests that the questions have not been resolved. A better sentence is: "The Staff's questions have been resolved and a discussion of the responses follows:

[redacted]

Just delete the specifics of the internal reference

internal ABB-CE correspondence [redacted] and further documented in the ABB-CENP System 80+ QA Plan, 18386-QO-001. In particular, ABB-CE design control procedures QPI 0304, "Design Analysis" and QPI 0306, "Design Verification" have been deferred. These procedures govern the methodology to document design analyses and the performance of design verification activities. The providers of engineering services (i.e., Stone and Webster Engineering Corporation, Duke Engineering Services, Inc., and ABB-Impell) to ABB-CE have performed independent design verification on portions of their design.

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

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

While ABB-CE has not conducted the formal independent design verification to date, the staff has concluded that a level of reasonable assurance for the integrity of the System 80+ design process has been obtained through:

- An evaluation of the original "ABB-CE Operating Experience Review for System 80+ MMI Design" was completed using the HFE PRM criteria. Overall, the ABB-CE operating experience review (OER) was quite impressive and showed a detailed review of many aspects of pertinent commercial nuclear power plant experience, and the subsequent incorporation of appropriate design features into the System 80+ design. Not all aspects of the HFE PRM were completely addressed however, and so ABB-CE worked with the staff to address the identified concerns. ABB-CE performed additional reviews of areas of operating experience resulting in additional items being added to the HF issues tracking system for later Consideration of incorporation into the design, at the appropriate time in the design process. This additional work is discussed in a revised OER, which also better describes how ABB-CE had already incorporated operating experience into the System 80+ design. The revised ABB-CE OER meets the criteria in the HFE PRM.

Functional Requirements Analysis and Allocation

HFE PRM Element 3, "Function Requirements Analysis," and Element 4, Function Allocation, specified that an analysis of functional requirements and a structured and documented allocation of functions should be performed. ABB-CE has stated that full analyses of functional requirements and function allocation are not necessary because the System 80+ design is an evolution of the System 80 design that was previously reviewed and approved by the NRC and has an operating history (Palo Verde Units 1, 2, and 3). In addition, ABB-CE has stated that both the definition and allocation of functions for the System 80+ are largely unchanged from its predecessor, the System 80. The reviewers agreed, and the HFE PRM criteria were modified accordingly.

ABB-CE's "Human Factors Evaluation and Allocation of System 80+ Functions" describes the critical functions and the success paths that are responsible for satisfying the safety functions. Comparisons are made at a high level between the System 80 and the System 80+ and differences are noted. The ABB-CE document is a useful information source that describes the basic structure of the System 80+ plant, the operator role that results from this

Human-System Interface Design

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

Review issues related to Element 6 were addressed in three major reviews, each pertaining to separate phases of the HFE PRM Element 6 review: (1) standard design features; (2) design methods and general characteristics; (3) human factors engineering standards, guidelines, and bases (HFESGB). Each is briefly described below.

Standard Design Features

Standard elements

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

The seven design features addressed by this review were found to be consistent with HFE design principles and guidelines. Further, the HSI design appeared to adequately address SPDS criteria. In some cases specific concerns were identified that could not be resolved at this stage of the HSI design. ABB-CE

DSER issues are also addressed. These considerations were evaluated within the context of the main control room configuration, the presentation of information on controls and displays, and the layout of panels. Specific attention was given to the RCS panel and the remote shutdown panel.

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

- Data entry tasks
- Blocking and tagging tasks via the data processing system (DPS) and the DIAS of instrumented and non-instrumented components
- Operator established alarms
- Component control system (CCS) operator module

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

HFE Standards, Guidelines, and Bases

The HFESGB review provides an evaluation of ABB-CE's HFE design criteria used to identify, select, and design equipment to be operated/maintained/controlled by plant personnel with respect to accepted HF guidance and practices. The ABB-CE document primarily addressed by this review is "Human Factors Engineering Standards, Guidelines, and Bases for Nuplex 80+" (HFESGB). This review addressed the following issues related to the design guidelines provided in these documents: technical basis and validity, level of detail, guideline integration, and procedure for implementation.

Certified Design Description/Inspections, Tests, Analyses, and Acceptance Criteria

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

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

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

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

Conclusions

The staff has reviewed the HFE process described by ABB-CE in the CESSAR-DC and CESSAR-DC-referenced documents. Based on its review, the staff concludes that the ABB-CE HFE program is acceptable and will result in acceptable HSI designs for the main control room, remote shutdown system, and related

room

applicable HSIs. The basic design features of the System 80+ advanced control room were reviewed and found consistent with HF standards, guidelines, and principles, and acceptable for use in the control room. In addition, the staff concludes that the design commitments and the HFE ITAAC/DAC accurately summarize the minimum HFE requirements for an acceptable design and verification/validation of the main control room and remote shutdown system. All previously identified DSER issues have been adequately addressed and are resolved.

Please delete the DAC reference in all "HFE ITAAC/DAC" uses. We have called the MCR & RSR ITAAC and have not referred to them as DAC. FSER Section 14.3 does not identify HFE as one of the two System 80+ DAC areas. It is potentially confusing and misleading to the reader to use "DAC".

Note: This reference is used in numerous places in ch. 18. All should be changed but only this one is uniquely identified.

- Functional Requirements Analysis and Allocation

- Task Analysis

- Human-System Interface Design

- Plant and Emergency Operating Procedures

- Verification and Validation

- Certified Design Description/Inspections, Tests, Analyses, and Acceptance Criteria

Section 18.11 provides a summary of the evaluation findings and overall conclusions.

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

detailed impletations on all control room panels.

In SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," the staff proposed the use of DAC as an approach to the review of the System 80+ design. This was due to the fact

The two DAC related TP should be removed per the comment on Page 18-12

that detailed design information was unavailable for selected areas of rapidly changing technology, including HF aspects of the control room and remote shutdown station design.

No HFE DAC have been specified

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

No review points specified Only ITAAC.

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

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

design certification. Specific issues were identified as requiring resolution. The issues are identified in Table 18.1-1. The table indicates the section in the FSER in which the DSER issue is addressed.

18.1.2 Final Standard Safety Analysis Report Review

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

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

18.1.3 Development of Review Criteria

18.1.3.1 Objectives

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

E

their resolution can be found in FSER Sections 18.7 and 18.8, respectively. To summarize, ABB-CE has committed to include procedure development as COL Action Item 13.5.1, Plant Operating Procedure (POP) Development Plan in CESSAR-DC. ABB-CE will develop the technical information required to serve as a basis for the detailed procedure development as part of the HFE process and this information will be provided to the COL. This COL action item is found to be acceptable.

With respect to impact on validation, ABB-CE included in CESSAR DC, a requirement in COL Action Item 13.5.1, POP Development Plan, for the COL to perform a POP validation effort that demonstrates the acceptability of the completed procedures. CESSAR-DC Section 18.9.3 Validation, was then modified to break validation into two phases. Section 18.9.3.1, Design Validation, addresses validation of the entire HSI without final procedures. CESSAR-DC Section 18.9.3.2, Operating Ensemble Validation Plan, and ~~HFPP~~ HFE V&V Plan Section 6.3.4.4, Operating Ensemble Validation Activities, addresses the "final" validation of the HSI after the final procedures have been completed. Operating Ensemble Validation requirements are addressed in CESSAR-DC Section ~~18.3.1.2.~~ 18.9.3.2 This validation, which will be performed by the COL, will provide assurance that trained operators using final, plant-specific procedures in the as-built CR form an effective operating ensemble. This two-phased validation approach and associated COL action item are found to be acceptable.

Status: Resolved.

18.2.4 Findings

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

These references are listed in CESSAR-DC without letter No. reference to USNRC

- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, System 80+ Human Factors Documentation Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases," (NPX80-IC-DP-790-01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Operating Experience Review for System 80+ MMI Design," (NPX80-IC-RR790-01, Rev. 00), ABB-CE letter dated December 18, 1992.

- Reference ~~7~~ #4 of CESSAR-DC Section 18.~~10~~4, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Human Factors Program Plan for the System 80+ Standard Plant Design," (NPX80-IC-DP790-01, Rev. 01, December 8, 1992), ABB-CE letter dated December 18, 1992. e? e? e?

- ~~Reference 2 of CESSAR-DC Section 18.10~~ LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment "Control Complex Information System Bases for Nuplex 80+" (NPX80-IC-DB791-01, January 15, 1993), ABB-CE letter dated January 18, 1993. e?

- NRC Internal Memorandum, Request for HICB Review of System 80+ Design Features - I & C, June 16, 1993.

- NRC Internal Memorandum, Request for HICB Review of System 80+ Design Features - I & C, June 23, 1993.

- Nuclear Management and Resources Council (1991). "Guidelines for Industry Actions to Assess Shutdown Management" (NUMARC 92-106). Washington, D.C.

- Nuclear Safety Analysis Center (1981). "Verification and Validation for Safety Parameter Display Systems" (NSAC-39). Palo Alto, CA.

- Public meeting minutes from September 10 and 11, 1992.

- Public meeting minutes from April 19 through 21, 1993.

- local control stations, and
- review of System 80 experience.

The staff had noted that the OER should review some of the more recent documents on local control stations developed in the review of the HF generic issue on local control stations, and those noted in paragraph 18.4.2.5 below. A list of seven pertinent local control station documents was provided to ABB-CE by the staff, who reviewed them and documented the review in Appendix C of the revised OER. Some items were identified where design guidance ~~was~~ should be considered ~~needed~~ for the System 80+. These issues were entered into the HFE tracking system. Further, the ABB-CE OER design resolutions appeared to somewhat narrowly exclude local control stations and the remote shutdown panel from their scope (see paragraph 18.4.2.2 below for specifics.) The revised OER has modified the design resolution section of these items to include local control stations within their scope. The review of System 80 experience will be discussed under other topics below.

Status: Resolved.

18.3.3.2.2 Analysis Results Report

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

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

Section 3 of the ABB-CE OER contains the detailed results of the OER analysis. There are a considerable number of HF/HSI issues addressed. Based on a review of the initial OER, the OER was modified by ABB-CE. The following modifications are particularly noteworthy:

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.

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

Issue II.K.1.10 in NUREG-0933 addresses the need to improve procedures. By letter dated November 24, 1992 (Reference 2 of CESSAR-DC Section 18.10, LD-92-115), ABB-CE indicated that this issue is not within the scope of design certification and will be a COL action item as noted in CESSAR-DC Section 13.5.2, "Administrative Control Procedures." This COL action item is found to be acceptable.

<u>DSER Issue 20.2-29: Nine Human Factors-Related GSIs</u>

1. <u>GSI Issue HF1.3.4.c (Man-Machine Interface (MMI) - Operational Aids)</u>: By letter dated December 18, 1992 (Reference 3 of CESSAR-DC Section 18.10, LD-92-120), ABB-CE provided information regarding this issue. ABB-CE indicated that the System 80+ MMI employs operator aids primarily to process data prior to presentation to the CR operators. The aids are integrated into the presentation hierarchy through application programs of the DPS and the discrete indication and alarm system (DIAS). The following operat~~or~~ional aids are provided as part of the System 80+ MMI (with the corresponding CESSAR-DC sections indicated):

- Signal reduction and validation - 18.7.1.4 and 18.7.3.2.1.6.
- Integrated Process Status Overview - 18.7.1.2.
- Alarm handling - 18.7.1.5 ~~and 18.7.2.3.~~
- Critical function monitoring - 18.7.1.8.2 and 7.7.1.10.
- Success path monitoring - 18.7.1.8.2.
- Core limit monitoring - 7.7.1.8.1.
- Computer aided surveillance testing - 7.7.1.8.2.M.

8. 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. ABB-CE indicated that all System 80+ local control stations are designed in accordance with the criteria in the System 80+ human factors engineering standards, guidelines, and bases. Further, ABB-CE noted that local control stations required to perform the System 80+ emergency operations guidelines are designed using task analysis and HF V&V. The staff finds this information acceptable and, therefore, this issue is resolved.

Nuplex 80+ Information Presentat

9. GSI Issue HF1.3.4.b (Annunciators): ABB-CE describes the System 80+ annunciator design in CESSAR-DC Sections 18.7.1.4, "~~Alarm Philosophy~~," and 18.7.1.5, "Alarm Characteristics." In CESSAR-DC Section 18.7.1.5, ABB-CE indicates that priority 1, 2, and 3 alarms are processed and displayed independently through both the discrete indication and alarm system (DIAS) and the DPS. The staff reviewed the relevant design documentation and conducted an onsite evaluation of the mockup of the DIAS and DPS, including the annunciator alarm system. A discussion of the staff's onsite evaluation is provided in Section 18.6.1.2.1 of the FSER. Results of the staff's review of the System 80+ alarm system relevant to Issue HF1.3.4.b are provided below.

One characteristic of the DPS is as follows: The DPS display hierarchy provides access to displays incorporating system/component status, process parameters, and annunciator status/acknowledgement. ABB-CE demonstrated available portions of the DPS display hierarchy on the mockup, including display navigation paths based on plant CSFs and plant segments and the representation of process parameters and system/component status via DPS displays. Also demonstrated were the incorporation of alarm status representations into these displays and the alarm acknowledgement capability. The incorporation of the alarms into the plant displays provides the capability to access alarm condition information and then acknowledge alarms from any DPS CRT in the CR. This characteristic provides flexibility to control room operations. IEC 964 (1.4-1) states, "An alarm shall be annunciated in the CR section where the operator has the necessary means for initiating corrective

18.4.5.1 ABB-CE Documentation

The following ABB-CE document was reviewed:

- Reference 7 of CESSAR-DC Section 18.4, LD-93-056, "System 80+ Human Factors Engineering," Attached, "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX-IC-RR790-02), Rev. 01, March 15, 1993), ABB-CE letter dated March 26, 1993.

The following documents were consulted as a part of this evaluation:

- September 10 and 11, 1992, public meeting minutes, dated October 21, 1992.

- Reference 7 of CESSAR-DC Section 18.4, LD-93-036, "System 80+ Human Factors Engineering Issue Closeout," Attached, "Human Factors Evaluation and Allocation of System 80+ Functions" (NPX-IC-RR790-02, Rev. 00, February 23, 1993), ABB-CE letter dated March 4, 1993).

18.4.5.2 DSER Review

18.4.5.2.1 DSER Issues

The staff's initial review of the this element concluded that ABB-CE had not confirmed that trade-off studies have been conducted to determine adequate configurations of personnel and system-performed functions. DSER Issue 18.6, "Function Allocation," was identified. At the September 10 and 11, 1992, public meeting, ABB-CE agreed to address the DSER issue in the document, "Human Factors Evaluation and Allocation of System 80+ Functions" by:

(a) describing the baseline system, its function allocation, and the changes and additions to function allocation for the new system,

(b) confirming that tradeoff studies or other means have been used to determine adequate configurations of personnel and system performed functions,

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attached, "Human Factors Program Plan for the System 80+ Standard Plant Design" (NPX80-IC-DP790-01, Rev. 01, December 15, 1992), ABB-CE letter dated December 18, 1992.

- Letter from T.V. Wambach (NRC) to ABB-CE, "Public Meeting September 10 and 11, 1992, Regarding Human Factors Engineering (HFE) Design Issues," (Docket No. 52-002).

- Public meeting minutes from September 10 and 11, 1992.

- Rasmussen, J. (1986). Information processing and human-machine interaction: An approach to cognitive engineering. New York: Elsevier Science (North-Holland).

- Time Response Design Criteria for Safety-Related Operator Action, American National Standards Institute, ANSI/ANS 58.8.

18.5.3 Evaluation of Element 5 - Task Analysis

The following review is organized in three major sections:

- Section 18.5.3.1 - Task Analysis Scope.

- Section 18.5.3.2 - Task Analysis Methodology.

- Section 18.5.3.3 - Issues Deferred from Element 4.

In the staff's initial review, DSER Issue 18.7 was defined. It provided criteria for defining the behavioral requirements of the tasks that personnel are required to perform to achieve the functions allocated to them. At the September 10 and 11, 1992, public meeting, ABB-CE agreed to update its task analysis methodology and incorporate the following modifications into Section 18.5 of CESSAR-DC:

Note, response to this DSER issue also addresses HFE PRM Element 5 Criterion 3, which requires that detailed task descriptions address staffing requirements including the number of personnel, their technical specialty, and specific skills.

Evaluation: ABB-CE defines staffing requirements including number of personnel and their technical specialties in Sections 18.3.2 and 18.6.2.2 of CESSAR-DC. ABB-CE's SSARFTA ensures that the HSI supports the operator's input and output requirements and that individual task elements are within human response capabilities. Coordination of activities between crew members is addressed by verification. Together, task analysis and verification adequately address these DSER and HFE PRM issues.

Status: Resolved.

Item I

Criterion: Provide commitment to address the following issues:

- maintenance work order tracking and tag out scheme for CR instruments and equipment identified via CRT and flat panel displays

- an account of how operators will track the status of equipment under test, surveillance, or repair

- impact of tracking scheme/system on normal, abnormal, and emergency operations

These items are addressed below.

I.1 Maintenance work order tracking and tag out scheme for CR instruments and equipment identified via CRT and flat panel displays.

Section 18.10

Evaluation. In Reference 7 of CESSAR-DC, LD-93-005, ABB-CE provided the following justification:

Tasks relating to maintenance work order tracking and tagout are not in the task analysis (both old and revised methods) because these tasks will not be performed in the controlling workspace and have no impact on the control room HSI design. A separate facility to support maintenance work tag-out is provided adjacent to the main control room in the System 80+ design.

This justification was found to be acceptable.

Status: Resolved.

I.2 An account of how operators will track the status of equipment under test, surveillance, or repair.

Evaluation: ABB-CE's response indicated that equipment status data will be inputted and maintained by personnel other than the operators. The operators will monitor the status of plant components and success paths through the DPS displays and the success path monitoring capabilities. The HSI characteristics for inputting this data was addressed by the Element 6 review of HSI design methods and general characteristics. ABB-CE provided a commitment to describe in greater detail the HSI for inputting these data.

Status: Resolved.

I.3 Impact of tracking scheme/system on normal, abnormal, and emergency operations.

Evaluation: ABB-CE's response indicated that the entry of and maintenance of status information will be performed by personnel other than the operators and the impact of unavailable components on safety and non-safety success paths will be determined by the DPS success path monitoring algorithms and indicated with alarms. In addition, CESSAR-DC Section ~~18.7.2~~ indicates that unavailable status of plant components will be indicated to the operator through various

18.7.1.8.2

Evaluation: Although the SSARFTA was not considered to be highly iterative by nature, it was acceptable with respect to the detailed requirements it provided for HSI elements: device type, measurement units, and value range, accuracy, and precision.

Status: Resolved.

18.5.3.3 Issues Deferred from Element 4

The following is a review of HFE PRM criteria from Element 4 that were deferred until Element 5 - Task Analysis.

<u>HFE PRM Element 4: Criteria 2, 3, & 5</u>.

Unchanged functions (Criterion 2) and modified functions (Criterion 3) shall be analyzed in terms of resulting human performance requirements. The results of analyses and trade-off studies shall support the adequate configurations of personnel- and system-performed functions (Criterion 5). ABB-CE was requested to describe how these issues will be addressed.

Evaluation: ABB-CE indicated (TA review issue A.1) that the necessary uses of new and modified functions (i.e., rapid depressurization, hydrogen ignitors, alternate generator, startup feedwater system) are specified in the procedure guidelines and operating sequences employed in the task analysis. ABB-CE stated further:

> The analytic scope of the TA [SSARFTA] will exercise the new and modified functions, extend the specified details of the operators' role from the function to the task level, identify human task performance requirements, and assess the resulting task loadings. Excessive loadings will result in further evaluation and formal resolution of the resulting allocation and design issues.

not stated in CESSAR ?

Status: Resolved.

10. Qualitative and quantitative criteria that identify when the operator is receiving "enough" rather than "too many" or "too few" number of alarms and displays

11. Auditable documentation to track the data/information that was lost/gained between System 80 and System 80+ control room designs

12. Effects (positive and negative) on operators performance of the changes, individually and collectively, between System 80 and System 80+.

The sub-issues and sub-items of Issue 18.8 were individually evaluated and resolved during relevant phases of the Element 6 review. Table 18.6-1 provides a cross reference between the DSER sub-issues/sub-items and the sections of the FSER where they are addressed.

18.6.1 Standard Design Features

The following is a review of important elements of the System 80+ design including six standard design features and the IPSO. A complete list of characteristics for each standard feature is provided in CESSAR-DC Section 18.7.1. Also discussed are DSER issues related to the HSI. Further, the CR design is reviewed against the staff's criteria for a SPDS.

18.6.1.1 Objectives

The objective of this review was to determine the acceptability of the basic design features of the System 80+ advanced CR on the basis of their consistency with established human factors standards, guidelines, and principles. The focus of this review was on the acceptability of these features as design concepts, as described in the CESSAR-DC and other design basis documents and as represented in the mockups of the MCC and IPSO.

elements

Reference not consistent with CESSAR-DC Ch 18

- Reference ~~3~~ 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment ~~4~~ 1, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992.

- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX-IC-DP-790-01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.

- Reference 1 of CESSAR-DC Section 18.10, LD-92-076, "System 80+ Shutdown Risk Report, Revision 1," attached "System 80+ Shutdown Risk Evaluation Report" (DCTR 10, Draft, June 15, 1992), ABB-CE letter dated June 16, 1992.

18.6.1.2.3 Design Criteria Documents

The following materials were consulted as part of this evaluation:

stet

- Reference 1 (stet) of CESSAR-DC Section 18.6 (stet), "Human Factors Engineering Standards, Guidelines, and Bases for System 80+" (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).

- Advanced Human-System Interface Design Review Guideline, Draft NUREG/CR-5908.

- A Status Report Regarding Industry Implementation of Safety Parameter Display Systems, NUREG-1342.

- Closure of Issues from the Draft Safety Evaluation Report (DSER) for System 80+, Docket No. 52-002, Letter from B.A. Boger (NRC) to D.M. Crutchfield (NRC), June 14, 1993.

- Compilation of Alarm System Guidelines and Evaluation of Their Applicability to Hybrid and Advanced Control Rooms, Draft NUREG/CR-6105, October 1991.

- CCS switch configuration

- IPSO

The first six features are standard in the sense that their basic design will be applied to various panels in the CR. Associated with each standard feature was a set of design characteristics, which were described in Reference 10 of CESSAR-DC Section 18.10, ALWR-92-203. In addition, the IPSO was included in this review. The main control room configuration (MCRC) was not evaluated because the design of the individual panels that comprise it was incomplete.

This review focused on the design basis of the design features and their associated design characteristics. In addition, a limited review of design implementation details was conducted for selected parts of the RCS panel and the chemical and volume control (CVCS) panel.

The ABB-CE mockup of the CR was used in this review. This mockup consisted of selected panels of the MCC in a static representation as well as portions of the RCS and CVCS panels in a dynamic stimulated HSI mockup. This mockup was not driven by a plant simulation. In addition, a static representation of the IPSO was presented via a rear projection display device.

The RCS panel is divided into three functional groups: reactor coolant pumps (RCPs) on the left, the RCS ~~in~~ on the ~~center~~ right, and the reactor coolant seal and bleed system ~~on~~ in the ~~right~~ center. Only the functional group for the ~~center~~ right portion of the panel, the RCS portion, was mocked up. Regarding the design features themselves, the DIAS multiple parameter display, the CCS process controller display, and the CCS component controls were not functional on the RCS panel and were observed and operated on the CVCS panel. Additionally, a limited set of the CRT screens of the DPS were designed and not all features of the DIAS displays were fully operational on the mockup. Further, the portions of the design that were completed, have been reviewed by ABB-CE, but findings not yet implemented on the mockup were due to cost, time, and higher priorities. Thus, discrepancies with ABB-CE's, "Human Factors Engineering Standards, guidelines, and Bases for System 80+" (HFESGB, Reference 1 of CESSAR-DC Section 18.6), guidelines were still present in the mockup.

addresses the operator's need to maintain awareness of significant changes in plant conditions and the implications of these changes to plant safety and operating goals. The basic concept of the DPS display hierarchy was found to adequately support this need. Final acceptance of the DPS display hierarchy will depend upon the final design implementation. ABB-CE has provided a set of 11 design characteristics associated with the DPS display hierarchy. Additional characteristics may also be considered in the post-certification review of the final design. A review of the 11 characteristics identified by ABB-CE is provided below.

18.6.1.3.1.1.1 Review of DPS Display Hierarchy - Design Characteristics

An initial review of the 11 characteristics associated with the DPS display hierarchy found the following characteristics to be acceptable - 1, 4, 5, 6, 9, and 10 - based on their support of the operator's need to access and process information regarding plant conditions. Characteristic 1 addressed the fact that the DPS display hierarchy provides access to the total set of plant data, as opposed to the subset provided by the DIAS. Plant data is organized in the DPS according to This includes CSFs and critical success paths and presented in display pages using graphical display formats such as schematic to support operator needs, diagrams and bar charts. The display pages are organized in a three-level hierarchy with increasing levels of detail to support operator information needs for monitoring, control, and diagnosis. Each display page provides a menu window to support navigation through the display hierarchy (characteristic 4).

Characteristics 5 and 6 addressed the fact that alarms may be acknowledged from the relevant DPS screens and that all DPS screens can be accessed from any DPS CRT in the CR. After an alarm has been acknowledged, a message describing the alarm condition appears in the spatially-dedicated area (characteristic 4). These characteristics support the operator's tasks by providing necessary information when it is needed. The DPS display units can be read at the control panel (characteristic 9). Greater viewing distances are not necessary because the full set of DPS displays can be accessed from isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, ~~C~~ A and ~~D~~ B instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The category 2 variables are displayed on DIAS-N and DPS with power supplies from the non-safety instrument buses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the ~~X~~ C and ~~Y~~ D instrumentation bus. The redundant information systems conform to the guidelines for the physical independence of electrical systems in RG 1.75.

The staff is reasonably assured that the information systems important to safety conform to the requirements of GDC 13 for monitoring systems and variables over their anticipated range for normal operation, for anticipated operational occurrences, and for accident conditions. Further, conformance to GDC 13 and the applicable guidelines satisfies the requirements of GDC 19 with respect to information systems provided in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Status: Resolved.

18.6.1.3.1.1.2 Review of DPS Display Hierarchy - Design Implementation

Selected portions of the DPS were reviewed using HFE guidelines and current design practice (subject to the detailed implementation limitations 'escribed above).

The following issues were identified.

Issue 1: DPS response time.

During the onsite evaluation, the time required by the DPS to respond to inputs was at times excessive. ABB-CE was requested to clarify the intended response time for the System 80+ and identify any response time differences between the design goal and the actual performance of the mockup.

The staff evaluated the plant process display instrumentation and has found the instrumentation to be acceptable. The staff's evaluation is presented in Section 7.5.2 of the DSER. The open items in Section 7.5 of the DSER have been resolved.

Status: Resolved.

Characteristic 12: Alarms are presented in one of four alarm states: new, existing, cleared, and reset.

Evaluation: The use of four alarm states is consistent with Guideline 3.2.2-1 of draft NUREG/CR-6105, which states that the annunciation sequence for each alarm should uniquely indicate: incoming alarms, acknowledged alarms, and cleared alarms. The use of coding schemes (flashing, ~~brightness~~ intensity, and audible tones) to designate these alarm states are generally consistent with the following guidelines from draft NUREG/CR-6105:

Note: changed to intensity See FSER R.6.3.2.3 p 3-49 CE changed terminology

- 3.2.2-1 (flashing, audible, and other visual coding)
- 3.2.2-2 (audible and other visual coding)
- 3.2.2-3 (audible coding)
- 3.2.2-4 (flashing and color coding)

The specific flash rates used in the implementation of codes for alarm state are a concern. ABB-CE indicated that new alarms would have a 1-second flash cycle with a 50-percent ON duty cycle (i.e., ON for 0.5 seconds and then OFF for 0.5 seconds) and the cleared alarms would have a 2-second cycle with a 25-percent ON duty cycle (i.e., ON for 0.5 seconds and then OFF for 1.5 seconds). The resulting flash rates for new and cleared alarms are 1 and 0.5 Hz, respectively. These flash rates are slower than the flash rates recommended by Guideline 1.3.10-13 of draft NUREG/CR-5908, which states, "A flash rate in the range of 2 to 5 Hz, with a minimum duty cycle (On interval) of 50 percent should be used." These flash rates are also slower than Guideline 2.3.3.3 of ABB-CE's HFESGB, which indicates that when two flash rates are used the higher priority state shall be between 3 and 5 Hz and the lower priority state shall be between 1 to 2 Hz. ABB-CE stated that it will implement flash rates that

- Post-accident monitoring instrumentation (PAMI) indication
- RG 1.97, "Instrumentation for LWR Nuclear Power Plants to Assess Plant and Environs During and Following Accident"

Evaluation: An effective method of supporting rapid comprehension of plant status is the use of spatially-dedicated, continuously-presented plant displays for key plant parameters. The CSFs, success path performance, RG 1.97/PAMI are key parameters that are related to safety. Due to their importance to plant safety they are acceptable parameters for the DIAS dedicated parameter displays. This is consistent with Guideline 1.1-22 of draft NUREG/CR-5908, which states, "Dedicated displays should be available to provide continuous indications of a minimum set of parameters necessary to assess the safety status of the plant."

The NRC HICB was requested to review this concern further. The staff evaluated the plant process display instrumentation and found the instrumentation to be acceptable. The staff's evaluation is presented in Section 7.5.2 of the DSER. The open items in Section 7.5 of the DSER have been resolved.

Status: Resolved.

Characteristic 12: DIAS dedicated parameter displays are diverse and independent of the DPS display system.

Evaluation: HICB was requested to review the above characteristic. The staff stated:

> The DPS is physically separated and independent of both DIAS channels. Independent Class 1E power busses are provided for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, ~~C~~ A and ~~D~~ B instrument busses. The DPS is powered from non-safety-related, battery-backed computer busses. The Category 2 variables are displayed on DIAS-N and DPS with power supplies from the non-safety-related instrument busses

and computer bus, respectively. Both are battery backed. The instrument channels are powered from the ~~X~~ C and ~~Y~~ D instrumentation bus. The redundant information systems conform to the guidelines [NRC staff guidance] for the physical independence of electrical system in RG 1.75.

Status: Resolved.

18.6.1.3.1.3.2 Review of Dedicated Parameter Display - Design Implementation

Selected DIAS dedicated parameter displays were examined and evaluated against HF guidance and current design practices. The following are issues that were identified.

Issue: Resolution of trend displays.

The vertical resolution of trend displays such as pressurizer pressure and level is quite small and may not be adequate for monitoring purposes. The normal operations band is narrow and the operator may have difficulty determining whether the current value is trending toward a limit. Discussions with ABB-CE during the onsite evaluation indicated that the range of the scales are not adjustable (i.e., operators cannot change the limits of the display such that a desired range of the scale is shown with greater resolution). ABB-CE was requested to describe its position regarding adjustable trend displays and any implications to other DIAS capabilities, such as the automatic range change features of the dedicated parameter displays.

Evaluation: During the onsite review, ABB-CE described its position regarding this display. ABB-CE stated that it is not intended that the DIAS displays be used to determine if values are trending towards limits. The DIAS control bands are only provided to be a reference point for where the normal operating range is. ABB-CE further explained that this display presents trends with the same resolution as the strip chart recorder that it replaces. These displays are provided to present continuously visible status histories for selected parameters that are controlled automatically. If the operator requires trends

Evaluation: HICB was requested to review the above characteristic. The staff stated:

> The DPS is physically separated and independent of both DIAS channels. Independent Class 1E power busses are provided for each redundant Category 1 sensor instrument channel, up to and including the channel isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, ~~C~~ A and ~~D~~ B instrument busses. The DPS is powered from non-safety-related, battery-backed computer busses. The Category 2 variable are displayed on DIAS-N and DPS with power supplies from the non-safety-related instrument busses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the ~~X~~ A and ~~Y~~ B instrumentation bus. The redundant information systems conform to the guidelines for the physical independence of electrical system in RG 1.75.

Status: Resolved.

18.6.1.3.1.4.2 Review of DIAS Multiple Parameter Display - Design Implementation

Selected DIAS multiple parameter displays were examined and evaluated against HF guidance and current design practices. The following are issues that were identified.

Issue: RCS flow indication.

The onsite review indicated that the parameter RCS flow is not available on the DPS or DIAS. The CSF walkthroughs indicated that this is a valuable parameter. ABB-CE is requested to state its position regarding the inclusion of RCS flow indications in the DIAS.

Evaluation: ABB-CE stated:

> The Function and Task Analysis shows no use of RCS flow, RCP differential pressure, SG differential pressure, or core differential pressure

the control panels. The following control modes were examined: master control level (e.g., RCS inventory), subloop level (e.g., charging), and component level (e.g., valve).

The CCS process controller display is an input device that combines the controllers for physically dissimilar but functionally-related systems into a single device that permits manual or automatic control at a number of different hierarchical levels (master, subloop, and component). This concept generally supports operator monitoring and control activities by organizing controls and displays by functional relationships and task requirements rather than by the physical relationships of plant equipment.

18.6.1.3.1.5.1 Review of the CCS Process Controller Display - Design Characteristics

An initial review of the 9 characteristics (see Section 18.7.1 of CESSAR-DC,) associated with the CCS process controller display found the following characteristics to be acceptable: 1, 3, 5, 6, 7, and 8.

Characteristic 6 addresses the HF concern that in traditional CRs operators must adjust and monitor numerous controllers that together achieve a higher-level control goal. For example, a set of separate, hardwired controls may be physically separated on the control panel by the type of plant system that is to be controlled (e.g., fluid system controllers may be separated from electrical system controllers). The CCS process controller display combines control interfaces for a variety of plant component/systems into a single interface. For example, controls for pressurizer heaters and spray are combined into a single controller. Characteristic ~~8~~ 7 addresses the fact that the CCS process controller provides the operator with three levels of control: master control, subloop control, and component control. Characteristic 9 addresses the fact Operator inputs at these levels include selection of manual or automatic control modes, selection of control signals for loop control, and selection of loop control set points. This configuration reduces the operator workload associated with monitoring and adjusting a dispersed set of controls.

ABB-CE stated that this is a prototype implementation issue that will be corrected by design review. The scale zero value will be marked to make it more salient.

Status: Resolved.

Issue 2: Deviation bar chart - normal control band.

Evaluation: The normal operating range is indicated by a vertical band along the scale. This band is thin and not highly salient. This makes comparison of the bar to the normal range difficult. This is a problem with the pressurizer pressure control resident on the RCS panel. ABB-CE stated that the CCS process controllers will not be implemented using that display unit but instead with the same type of display unit as the pressurizer level control of the CVCS panel. The deviation bar charts of the pressurizer level controller were found to be acceptable.

Status: Resolved.

HFESGB in section 2.4.3.a) states the following: "Interface Hardware devices shall be demarcated to not less than one-half the minimum precision required by the users tasks.

Issue 3: Deviation bar chart - scale resolution.

Evaluation: The deviation bar charts for charging and letdown have scale demarcations in units of 10, with a range of -20 to +20, while the actual values are presented with a resolution of a single unit (e.g., 0.01m^3/min (3 gpm). This appears to conflict with guidelines that state that the resolution of the scale should match the resolution requirements of the user's

?

information reqes a fairly deliberate action. Simply sliding one's finger across the screen should not activate a touch area.

ABB-CE stated that if an accidental activation did occur, the consequences would be no worse than if the component actuation were due to a single failure or push button switch misoperation (e.g., loss of electrical power). Additionally, ABB-CE stated that when specifically required by fluid system design, provisions to preclude inadvertent component actuation are provided in the CCS (e.g., component breaker rackout, administrative controls).

ABB-CE stated that it is not its policy to provide confirmatory messages for the CCS process controller displays because these messages do not effectively prevent errors of intent and the consequences of erroneous input are not immediate and severe.

Status: Resolved.

18.6.1.3.1.6 CCS Switch Configuration

The onsite review examined the CCS switch configuration for the CVCS makeup system on the CVCS panel. In addition, non-functioning mockups of the switches were observed at the RCS panel for control of reactor coolant pumps and backup pressurizer heaters. The switches consisted of physical push buttons with backlit legends that indicated operating status. Color coding was used to indicate status (e.g., red = active, green = inactive). In addition to status conditions, the component discrepancy state was examined. A component discrepancy occurs when the demanded state of a component (demanded by remote, automatic control action) is different from the actual state of the component. This condition was indicated by the concurrent flashing of both the red ~~and~~ or green backlit portions of the switch associated with the current state of the component

The CCS switches are spatially-dedicated, functionally-grouped input devices that allow the operator to provide discrete control inputs (as opposed to control set points for automatic controllers as with the CCS process controller). This conceptual design generally supported operator requirements for controlling components and monitoring component status. The design and placement of the switches was found to be acceptable based on consistency with HFE guidelines for coding, spatial dedication and functional grouping and on functional similarity to other existing control panel switches. The acceptability of specific characteristics of the switches are reviewed in the sections below.

The following parameters, identified in NUREG-1342 as important SPDS parameters, were either not present or not fully implemented on the IPSO mockup. These parameters are listed below according to the safety functions used by NUREG-1342:

- Reactivity control
  - Source range

- Reactor coolant system integrity
  - Steam generator pressure
  - Containment sump level

- Radioactivity control
  - Effluent stack radiation
  - Steamline radiation
  - Containment radiation

- Containment Conditions
  - Containment pressure
  - Containment isolation status

A justification for not needing SG pressure for CE plants per NUREG-1342 was provided. I believe SG pressure was removed from the list of parameters to be added.

See Page 18-185

Evaluation: In the April 19 through 21, 1993, meeting minutes, the NRC staff stated that this issue can be acceptably resolved by ABB-CE's commitment to incorporate these parameters, plus containment hydrogen concentration, into the IPSO display and to enter this commitment into its open issue tracking system. ABB-CE agreed to include these parameters and entered Item 62 into the TOI to record its commitment.

Status: Resolved.

Issue 2: Trend indication symbols for key parameters.

Evaluation: The IPSO display includes arrows beside the digital values of key parameters to indicate the direction of change. This concept is consistent with Guideline 1.3.5-6, Direction of Change in Digital Display, of draft NUREG/CR-5908.

stated that adequate overlap would be provided between the reactor vessel level and the refueling level instruments. Alarms related to these instruments are also not yet complete.

Thus, ABB-CE has taken appropriate actions to address the low-power and shutdown area, given the current stage of the System 80+ design, and continues to give this important area attention. It is anticipated that those areas that have not yet been designed will receive the similar attention by ABB-CE as the design process proceeds.

18.6.1.3.2.3 System 80 Experience

The third area examined for RCS-related OER items was the System 80 operator interviews. A number of items were selected from these interviews to review. These are summarized below, along with the status of the ABB-CE design in each area. Subsequent to the onsite reviews, ABB-CE provided a formal response to the System 80 operator interviews (Ref. 4 of CESSAR-DC Section 18.10, LD-93-135).

1. Tracking of heat up and cool down rates - ABB-CE has developed a proof of principle DPS screen to aid the operators in tracking these rates, but further work is needed. It will be finalized during the design process. An example of one area needing improvement is the cooldown rate, which as currently provided, is only based on a one-hour time frame. In addition to this one-hour-based rate, the operators need a rate that is based on a much shorter time interval for control purposes. ABB-CE stated that consideration would be given to that during the design process.

2. Operator decision aids to assist in initial post-trip actions - ABB-CE is considering such an aid but has not yet developed it. Item 41 has been entered into the TOI system to address this issue.

3. Exploration of automation of RCP seal isolation - ABB-CE has entered Item 86 42 in the TOI to record its commitment to evaluate the need for this and other operator decision aids.

the proximity of the alarm setpoint to a significant operator action condition. Alarms are organized into three levels of priority with priority 1 being the last warning prior to a significant operator action condition, priority 2 being the next to last warning, and priority 3 being any number of warnings prior to the next to last warning. The alarm priority scheme including six categories of significant operator actions were reviewed as part of the review of design methods and general characteristics and found to be acceptable. Alarm priority is represented by the following shape codes: priority 1 - an illuminated box, priority 2 - an illuminated frame, and priority 3 - illuminated brackets (four corners of the frame).

Alarm state has four levels (new, existing, cleared, and reset) that are coded by tile ~~brightness~~ intensity and flash rate. These are applied to the shape (e.g., ~~box~~ reverse video, ~~frame~~ box, or brackets) surrounding the alarm tile. New alarms have the ~~brightest~~ highest intensity shape and flash with a 50/50 on-off cycle. Existing alarms have an intermediate level of ~~brightness~~ intensity and do not flash. Cleared alarms have the lowest level of ~~brightness~~ intensity and flash with a 25/75 on-off cycle. Reset alarms have no illumination.

Note: CE changed terminology

See FSER 18.6.3.2.3 p 18-249

ABB-CE described a design process in which various design concepts for the alarm tiles were generated, subjectively evaluated, and modified. ABB-CE described informal experimentation and subjective evaluation of various alternatives for the alarm coding scheme. No formal process for collection and analysis of empir cal data was presented by ABB-CE. While the information coding schemes are consistent with general HF guidance for information coding, concerns were identified regarding the specific coding values that were implemented in the design. ABB-CE entered Items 74, 75, 76, and 78 into the TOI system to record its commitment to address these concerns.

While individual dimensions of the alarm coding scheme may be consistent with HF guidelines, the effectiveness of the overall alarm coding scheme, including the integration of shape, flash, and ~~brightness~~ intensity codes remains largely untested. ABB-CE entered Item 77 into the TOI to record its commitment to evaluate the effectiveness of the alarm system through verification and validation activities when the system is fully implemented. In addition, ABB-CE entered Item 101 into the TOI to record its commitment to evaluate the alarm system using a prototype of the DIAS alarm tile prior to verification and validation.

Based on these commitments from ABB-CE, it was recommended that DSER Issue 18.8.1.1 be closed.

Status: Resolved.

18.6.1.3.3.2 DSER Issue 18.8.1.3 - Flash Coding of Alarms

As a result of the meeting of September 10 and 11, 1992, ABB-CE committed to provide a rationale for the alarm flash duty cycle that is 50/50 on-off for new alarms and 25/75 on-off for cleared alarms. This rationale was to include a justification for inconsistency with NASA 3000, "NASA Man-Systems Integration Standards," (1989) which states "Flashing lights shall have approximately equal amounts of ON and OFF time."

Evaluation: The alarm system uses flashing as a coding scheme to draw attention to those changes in alarm states that require an acknowledgement from the operator (i.e., new and cleared alarms). Because multiple alarm conditions are associated with each alarm tile, the flash rate was configured to allow more than one alarm state to be conveyed. For example, the coding shape for a new alarm is visible during the ON portion of its duty cycle and the coding shape for an existing alarm may be visible during the OFF portion of the new alarm's duty cycle. ABB-CE stated that the alarm system employs different flash rates for new and cleared alarms to compensate for the possibility that the flash rates for new and existing alarms may drift and overlap. If overlap did occur the new alarm would not be masked by the cleared alarm; the new alarm would be visible because it has a longer ON cycle than the cleared alarm. ABB-CE further stated that flash coding is redundant with ~~brightness~~ intensity coding and that similar duty cycles are used for new and cleared alarms in traditional CRs that have tile annunciators.

See note on p 18-178

The rationale for the flash rates is satisfactory. However, because the effectiveness of the overall alarm coding scheme, including the integration of

18.6.2.2.2 Material Reviewed

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

- Reference 11 of CESSAR-DC Section 18.10, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 1 (untitled), attached response to RAI No. 620.2, ABB-CE letter dated May 8, 1992.

- Reference 6 of CESSAR-DC Section 18.10, LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX80-IC-DP-790-01, Rev. 00, January 15, 1990), ABB-CE letter dated September 23, 1992.

- Reference 3 of CESSAR-DC Section 18.10, LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), attached response to DSER Item 20.2-29," ABB-CE letter dated December 18, 1992.

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18 DSER Open Item Responses," ABB-CE letter dated January 18, 1993.

- Reference 12 of CESSAR-DC Section 18.10, LD-93-135, "System 80+ Information for Issue Closure," Attachment 6, Sub-Attachment 1, "Comments from Draft TER (July 14, 1993) on Nuplex 80+ HSI Justification of ABB-CE Positions Requested for Closure of HSI Issues," ABB-CE letter dated September 1, 1993.

- Reference 13 of CESSAR-DC Section 18.10, LD-93-147, "System 80+ Information for Issue Closure," Attachment 1, "Response to Cross-Branch Chapter 19 Questions (October 4, 1991)," ABB-CE letter dated October 18, 1993.

- Reference ~~1~~ 6 of CESSAR-DC Section 18.~~6~~ 4, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+," (NPX80-IC-DR-791-02, Rev. 00), September 15, 1993).

Design Issues Related to Control Room Configuration Design

The following issues are relevant to Criterion 1 as well as other criteria of HFE PRM Element 6. ABB-CE's responses to these issues are discussed. The full text of ABB-CE's responses may be found in Reference 14 of CESSAR-DC Section 18.10, LD-93-135.

1. Issue: Section 18.6.6.1 of CESSAR-DC states a minimum CR ventilation rate of 0.42 cubic m (15 cubic ft) of air per minute. NUREG-0700 states a minimum of 0.42 cubic m (15 cubic ft) per minute per room occupant. Since a supervisor may be frequently present in the CR, in addition to two operators, the minimum ventilation value stated by ABB-CE appears to be too low. ABB-CE was requested to address this apparent discrepancy.

Evaluation: In its response, ABB-CE stated that Guideline 7.3.1 of HFESGB correctly cites the NUREG-0700 guideline regarding ventilation, ~~and that~~ Section 18.6.6.1 of CESSAR-DC ~~will be~~ was modified to conform to NUREG-0700. indicate that Control Room ventilation rates will meet HFESGB guidelines.

Status: Resolved.

2. Issue: Section 18.6.6.1 of CESSAR-DC states that background noise levels will be in accordance with HFESGB, which states a maximum background noise level of 65 db(A) and a reverberation time of one second or less. This section also states that workstation lighting will be in accordance with HFESGB, which states detailed illumination criteria. ABB-CE was requested to specify how the environmental conditions of the CR will be evaluated. Will they be verified by ABB-CE in an ABB-CE facility or will this be a COL responsibility? What tools and methodology will be used for t[illegible] evaluation.

Evaluation: ABB-CE stated that workspace (e.g., CR) environmental conditions will be evaluated through survey and measurement of the actual as-built facilities per applicable criteria from the HFESGB. This is part of the suitability inspection specified in the HF V&V plan, which in turn is part of the verification of suitability required by the HFPP and ITAAC items for the MCR and remote shutdown panel. Verification of environmental conditions is thus a COL applicant responsibility. Selection of personnel to perform the activity will be at COL applicant discretion. ABB-CE further stated that while the measurement (i.e., acceptance) criteria need to be specified, it is not necessary to specify measurement tools at this time.

Status: Resolved.

3. Issue: Section 18.6.5.6.1.2 of CESSAR-DC states that the MCC is visible from a central location at either the auxiliary console (AC) or safety console (SC). However, from Figure 18.6.5-8, it appears that only a portion of the plant monitoring and control panel, and none of the RCS and CVCS panels are visible from the AC and SC. ABB-CE was requested to clarify its statement including a discussion of the possible effects of impaired visibility of the MCC. ABB-CE was also asked to address the apparent discrepancy between these visibility limitations and the design requirement for large digital readouts on the DIAS displays.

Evaluation: ABB-CE stated that visibility of the MCC area from the AC and SC panels is provided to facilitate operator communication and coordination, not for direct monitoring or reading activities, and is therefore acceptable. In addition, the CRT displays on the AC and SC panels provide access to all information available at the MCC. IPSO provides plant overview information that can be read throughout the CR. DIAS digital displays are designed to be read across the MCC (e.g., read RCS panel DIAS displays while standing at the turbine panel), not across the CR. ABB-CE has agreed to revise CESSAR-DC Section 18.6.5.6.1 to clarify the statements that refer to visibility.

Status: Resolved.

4. Issue: Section 18.6.5.6.1.4 of CESSAR-DC states that unobstructed visual access exists to the MCC from the CR supervisor (CRS) and shift supervisor (SS) offices. However, from Figure 18.6.5-9, it appears that the RCS panel is not visible from the shift supervisor's office. ABB-CE was requested to address this apparent contradiction.

Evaluation: ABB-CE stated that visibility of the MCC area from the CRS and SS offices is provided for general observation. It is not intended to support direct personnel supervision or plant monitoring. ABB-CE agreed to revised CESSAR-DC Section 18.6.5.6.1 to clarify the terminology and related subordinate statements. (See also Issue 3 of Section 18.6.2.3.2.2.)

In Amendment V

Status: Resolved.

5. Issue: Section 18.3.2 of the CESSAR-DC discusses CR staffing and the design bases for the CR configuration. Section 18.6.1 defines the various terms used in describing the control room configuration, such as "controlling workspace," and "control room." 10 CFR Part 50 uses the terms "at the controls" and "control room." RG 1.114 provides guidance in detail as to what is meant by and necessary for these areas. One example is the need for an unobstructed view of controls, displays and alarms "at the controls." ABB-CE has not used the same terms and has not provided a commitment to RG 1.114, thus making it unclear as to their commitment to the detailed guidance and requirements of the RG and 10 CFR Part 50. ABB-CE was requested to provide such a commitment or alternatively describe clearly their method to be used in place of the RG. ABB-CE was also requested to provide one of their CR figures that clearly demarcates the pertinent areas.

Evaluation: ABB-CE stated:

> The design of the Nuplex 80+ control room will accommodate the COL applicant's meeting of the requirements of 10 CFR 50.54(k) and (m), and RG 1.114. However, RG 1.114 presents behavioral and administrative requirements on COL applicant operators, rather than design requirements; thus it does not form the basis for a coherent commitment by ABB-CE. Compliance with these issues, as RG 1.114 states, are COL applicant responsibilities.
>
> Nonetheless, the Nuplex 80+ philosophy and design are cognizant of and consistent with the general intent of RG 1.114, i.e., to keep

4. Issue: Section 18.7.1.1.~~2~~ 8 of CESSAR-DC indicates that ~~positions of non-instrumented valves~~ component or parameter information unavailable from automated data acquisition means are entered ~~by procedure~~ manually into the DPS. ABB-CE was requested to describe the human-computer interface that will be used for entering data including data entry screens, methods of interaction, provisions that will be made to reduce input errors, and provisions that will be made to ensure that entered values are kept current.

Evaluation: ABB-CE stated that a controlled-access interface for data entry (i.e., a keyboard) has been functionally specified. The interface will be located on the CRS console and in MCR office(s). An item has been entered in ABB-CE's HF issue tracking system (Item 95) to ensure treatment of this human-computer interface in subsequent design and V&V activities.

Status: Resolved.

5. Issue: Section 18.7.1.1.4 of CESSAR-DC describes the assignment of alarms into categories (e.g., priorities 1 to 3 plus a fourth category called operator aids). This assignment is based on the proximity of the alarm setpoint to the significant operator action conditions. The following issues were identified.

a. The meaning of the term "significant operator action" should be defined. This definition should include the implications for automatic system actuations. For example, if a condition will result in the activation of an automatic protection system (e.g., safety injection or reactor trip) no operator action may be required. In this case is the alarm considered to be high priority?

Evaluation: ABB-CE stated, "Significant Operator Actions are those judged to be necessary to prevent specific undesirable consequences; these will often be redundant with automatic (i.e., protective) actions (for defense in depth). Alarms are not associated with automatic actions, per se."

Status: Resolved.

d. ABB-CE indicated that additional rules may have to be added before the alarm categorization is complete. ABB-CE was requested to document these rules in their design documentation and to keep these records current.

Evaluation: ABB-CE stated that any added or revised rules will be incorporated in the alarm system design documentation.

Status: Resolved.

6. Issue: Section 18.7.1.1.7 of CESSAR-DC specifies that the IPSO shall be readable from the shift supervisor's office. ABB-CE was requested to identify the criterion that will be applied for legibility of the IPSO from the shift supervisor's office.

was

Evaluation: ABB-CE stated the criterion of 15 minutes of arc (MOA) for minimum character height found in Section 18.7.1.1.7 of CESSAR-DC was outdated and will be revised to indicate a minimum height of 12 MOA for any specified reading distance. This will make CESSAR-DC consistent with HFESGB Section 2.2.3.2.b. Letter heights on the IPSO (5 cm (2.1 in.)) at the specified reading distance from the SS office (approximately 12m (40 ft)) yield a proximal character height of 15 MOA. Since these values are ultimately based on the position of the reader with respect to the display, and since there are no tasks outside the controlling workspace that preclude viewers from adjusting their position for a better view of IPSO, ABB-CE considered the present character sizes to be acceptable.

The acceptability of ABB-CE's legibility criterion of 12 MOA, rather than a value of 15 MOA as specified by NUREG-0700 or 16 MOA as specified by ANSI-HFS-100, was addressed by the HFESGB review. ABB-CE's rationale for using 12 MOA as a robust criterion for legibility was not found to be fully supported by HF literature. However, it was acknowledged that actual viewing distances may be less than design assumptions stated in HFESGB. ABB-CE agreed to include Item 102 in its HF issue tracking system to ensure that legibility be further evaluated under conditions that are representative of anticipated work conditions. The character height of 5 cm (2.1 in.) for the IPSO was found acceptable on the basis that it resulted in a visual angle of 15 MOA, not 12 MOA, when viewed from the SS office.

Status: Resolved.

7. Issue: Section 18.7.1.1.7 of CESSAR-DC states that a DPS CRT located on an adjacent panel may be used by the operator to support monitoring tasks. CESSAR-DC, Amendment E stated that data on CRTs is sized for readability assuming the largest CR panel size. In CESSAR-DC Amendment N, that sentence was omitted. ABB-CE was requested to describe the degree of legibility (e.g., only alarm symbols versus all text) required of CRTs on adjacent panels to support operator task requirements. ABB-CE was also requested to describe how the DPS screens will be designed (including design criteria for character heights) to address these viewing requirements. space

Evaluation: ABB-CE stated:

> Although DPS screens do not normally need to be read from adjacent panels, the specified DPS screen character size (4 mm (.175 in.)) yields a proximal character height of 12 MOA at the specified reading distance (127 cm (50 in.)). This value is sufficient between panel centers, and is reasonably robust to off-angle viewing (see basis for HFESGB Section 2.2.3.2.b). The 12 MOA value also meets the criterion of NUREG-0700 Section 6.7.2.2.b(1) for character size on CRT displays.

Verification of legibility of CRTs from adjacent panels is addressed by HF issue tracking Item 102.

Status: Resolved.

8. Issue: Section 18.7.1.6.2 of CESSAR-DC discusses various mechanisms for controlling and indicating components and systems. Items addressed include: control location, engineering safety features actuation system

[physical] covers is not necessary or anticipated. The equivalent treatment for process controllers has not yet been determined. Input of not-instrumented component status information is intrinsically an administrative control issue. On the other hand, how control blocking will be implemented in the CCS is a design issue and has yet to be addressed. ABB-CE acknowledges this to be an important set of issues; their detailed treatment has been entered as TOI Item 96 for future treatment.

Status: Resolved.

11. Issue: A review of the design documents and the HSI mockup indicates the apparent lack of a systematic approach for determining the degree of precision with which data are presented, whether in digital or graphic form, to the operator via the HSI. Section 18.5.1.5.~~X~~3 of CESSAR-DC states that parametric requirements for display and control variables will be defined in terms of device type, range, accuracy, and units as part of the FTA methodology. While accuracy of data is an important requirement, it is a separate concern from the precision with which data is presented to the operator (e.g., the number of significant digits in digital displays, the number of intervals on scale displays). Specific examples of the lack of clearly defined display precision requirements were observed during the design features review with respect to the scaling on bar charts and other indicators. The ABB-CE HFESGB document provides general criteria for scaling but this is insufficient for determining the precision requirements for specific parameters. ABB-CE was requested to define a systematic process by which precision requirements will be defined for displayed values.

Evaluation: ABB-CE agreed to enhance the guidance for specifying precision that is contained in the HFESGB Section 2.4.3. ABB-CE agreed to modify modified Section 18.5.1.5.3 - Information and Control Requirements of CESSAR-DC to is a time history plot of the most recent data over a specified short duration (e.g., 30 min). The purpose of a trend display is to explicitly present a timeserial view of the parameter's recent and ongoing changes. This supports extraction of higher order information (i.e., first and second derivatives), observation of process characteristics, and the extrapolation and prediction of future process values. Trend data is retained during display, not retrieved prior to display. TOI database entry 91 has been made to ensure that the HSI for the HDSR is provided and designed in accordance with HFPP requirements, and that the HDSR parameters to be stored and the data/display resolutions are defined.

Status: Resolved.

18.7.1.5.6.D

9. Issue: Sections 18.7.3.2.3.~~3~~4 and 18.7.1.5.~~5~~2 of CESSAR-DC describe priority 2 operator established alarms. Two concerns exist: alarm establishment and alarm presentation. Section 18.7.3.2.3.3 briefly e describes the process by which the operator may establish new alarms, which includes accessing a database and entering new alarm setpoints. ABB-CE was requested to describe the interface to be used to perform this task including displays to be accessed and input devices used to supply setpoints and applicable alarm messages. With respect to the representation of operator established alarms, ABB-CE was requested to describe measures that will be taken to ensure that operator established alarms are not confused with each other or with standard plant-generated alarms. In addition, ABB-CE was requested to describe constraints on the number of parameters and the number of setpoints per parameter for operator established alarms and how operator established alarms will be managed across shift turnovers.

Evaluation: ABB-CE indicated that the operator-established alarms have a dedicated alarm tile on each panel and each operator established alarm has a separate alarm message. However, the design details of the interface for operator established alarms are not yet completed. Item 87 has been entered into ABB-CE's HF issue tracking system to ensure that the identified concerns are addressed.

Status: Resolved.

10. Issue: Section 18.7.3.2.3.5 of CESSAR-DC states that priority 3 alarms are only available on the DPS and individual alarm tiles are not required for these conditions. Other sections of CESSAR-DC and the onsite review have demonstrated the use of priority 3 alarms on the DIAS alarm tile display. ABB-CE was requested to clarify this apparent contradiction.

Evaluation: ABB-CE stated:

> Priority 3 parameters that do not degrade to priority 2 or 1 conditions are processed and displayed only by the DPS. DPS performs processing and display of all alarms and operator aids. CESSAR-DC Section 18.7.3.2.3.5 will be clarified with regards to the DIAS alarm system. The final RCS panel design will be modified to incorporate the System 80+ standard features and conventions described in other sections of CESSAR-DC.

In Amendment was

Status: Resolved.

11. Issue: Section 18.7.3.6 of CESSAR-DC states that an operator aid alarm tile is provided in the lower-right corner of the DIAS alarm tile display. Several issues are described below.

a. Figures 18.7.3-39 apparently identifies this as a tile for operator established alarms, not an operator aid. Section 18.7.1.5.5 states that operator aids are only presented on the DPS. Is this tile actually an operator established alarm tile?

Evaluation: ABB-CE stated that this tile is actually an operator established alarm tile. Operator aids are only presented on the CRTs, as stated in CESSAR-DC Section 18.7.3.2.3.6.

Status: Resolved.

b. How many operator established alarm tiles will be provided per DIAS alarm tile display and per panel of the MCC?

Evaluation: ABB-CE stated that only one operator established alarm tile per panel will be provided. ~~ABB-CE stated that~~ CESSAR-DC Section 18.7.3.2.3.4 ~~will be~~ was revised to clarify this.

Status: Resolved.

c. How many plant parameters may be associated with a single operator established alarm tile?

Evaluation: ABB-CE stated that this is a design detail and is addressed by Item 87 of its HF issue tracking system as described in Issue 9 of this section.

Status: Resolved.

d. How may setpoints may be associated with a single parameter of an operator-established alarm tile?

Evaluation: ABB-CE stated that this is a design detail and is addressed by Item 87 of its HF issue tracking system as described in Issue 9 of this section.

Status: Resolved.

12. Issue: Section 18.7.1.5.5 of CESSAR-DC describes operator aids, as information that is helpful to the operator for plant control, but lower in priority than priority 3 alarms. Operator aid information will be

presented on the DPS CRTs using an ". . .orange underline of the text of the information it applies to. The operator aid information flashes when unacknowledged and then may be acknowledged by the operator; however, there is no reset state." The following concerns were identified.

a. The content and appearance of the operator aid should be described in greater detail. For example, where will the text reside (e.g, in the message window?, in the main part of the screen?).

Evaluation: ABB-CE stated that the operator aid text will reside in the message window on the lower part of the CRT screen. ABB-CE has entered Item 100 into its HF issue tracking system to ensure that it will provides an operator aid illustration in the future following further implementation of operator aids in the prototype.

Status: Resolved.

b. The coding scheme, which was an orange underline, appears to conflict with Section 18.7.1.1.2 of CESSAR-DC which states that the color white will be used for operator aids and orange will be used for operator established (alarm) information.

was

Evaluation: ABB-CE stated that an orange underline is used for operator aids. ABB-CE also stated that CESSAR-DC Section 18.7.1.1.2.G will be corrected to say "operator aids" instead of "operator established information."

Status: Resolved.

13. Issue: Apparent inconsistencies were noted within the DPS with respect to abbreviations. For example, the DPS display, "Inventory Control (CFM) Level 2," shown in Figure 18.7.1-6 of CESSAR-DC provides a poke area labeled "PZR PRES" for quick access to a supporting diagnostic page. However, the corresponding designator of the PRI menu page shown in the Figure 18.7.1.5 of CESSAR-DC is labeled "PZR PRESS". Other apparent inconsistencies were noted with the use of the abbreviations SI and SIS within the IPSO display and the rest of DPS display hierarchy. These

4. Issue: The message tile monitor shown on Figure 18.7.4.3 for the feedwater and condensate system is not described. Is this actually a DIAS alarm tile display?

Evaluation: ABB-CE stated that the message tile monitor shown on Figure 18.7.4.3 was actually a DIAS alarm tile display that was incorrectly labelled.

Status: Resolved.

5. Issue: Section 18.7.4.5 of CESSAR-DC states that the CCS module of the safety monitoring panel provides access to all CCS controls and indications. (This capability is also stated for the remote shutdown panel.) The number of controls that may be accessed through the module may be large and impose high demands on the operator for control access and status monitoring. ABB-CE is requested to describe this module in greater detail along with provisions for facilitating control access and status monitoring.

Evaluation: ABB-CE stated that the design for the operator's module is not complete. ABB-CE has entered Item 97 into its HF issue tracking system to ensure that the demands on the operator for control access and status monitoring are addressed during suitability analysis.

Status: Resolved.

6. Issue: Section 18.7.4.14 of CESSAR-DC describes the control room supervisor's (CRS) console. The following issues were identified.

- The method by which work space requirements were identified is not described. ABB-CE was requested to describe the basis for the proposed design.

- Two potential benefits of including two DPS terminals in the CRS console are to compensate for the absence of dedicated (DIAS) indications and to allow rapid cross-checks to be made between different DPS display pages.

demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options.

Evaluation: Sections 18.7.2.4 and 18.8.1.2 of CESSAR-DC state that the remote shutdown panel design uses the same panel profile as the main control console. It also uses the same criteria for human engineering and for information display and control allocation as the MCR. A review of criteria for selection and design of HSI hardware and software approaches are addressed in the review of MCR.

Status: Resolved.

4. Criterion: HFE PRM General Criterion 7 states that the HFE standards shall be employed in HSI selection and design. Staffing assumptions for the remote shutdown panel and the local control stations are important considerations for the application of these criteria. ABB-CE was requested to describe its staffing assumptions for the remote shutdown panel and the local control stations.

Evaluation: ABB-CE stated that staffing assumptions for the remote shutdown panel and the local control stations will be provided on a task-specific basis via FTA, consistent with the 10 CFR 50.54 and RG 1.114 staffing requirements.

Status: Resolved.

5. Criterion: HFE PRM General Criterion 9 states that HFE shall be applied to the design of equipment and software for maintainability, testing, and inspection. In particular, the following are not clearly described: (1) provisions for maintenance at locations in the plant such as local control stations; and (2) provisions for in-service, surveillance testing in the remote shutdown panel, local control stations, and other locations in the plant.

success path applications. However, the fact that dedicated tiles are not provided does not mean that the alarms are not available. All alarms are available on the DPS at the RSP.

Note → This table was removed and CESSAR-DC Section 18.8.1.4 was changed to indicate that RSP alarm requirements will be identified as part of the Functional Task Analysis and detailed panel design.

Status: Resolved.

3. Issue: Table 18.8-2 of CESSAR-DC states that the RSP will use a single RCP trouble alarm tile instead of the 16 RCP dedicated alarm tiles and the two seal/bleed alarm tiles that are provided on the RCS panel in the MCR. This appears to conflict with good design practice for alarms as reflected in EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development," which states, "Use of shared, or so-called "trouble" annunciator tiles should be minimized" and NUREG-0700 which states, "Annunciators with inputs from more than one plant parameter set point should be avoided." ABB-CE was requested to describe its rationale in greater detail. What are the implications for operator workload for processing alarms? How will this affect the operator's ability to rapidly determine the state of the RCPs when multiple alarm states have been tripped? What provisions have been made to ensure that the many alarm states associated with the RCPs will not interfere with the operator's ability to access other alarm information from the DIAS alarm tile display, including when using the alarm list displays.

Evaluation: ABB-CE stated:

The Nuplex 80+ tile reduction philosophy acknowledges the paradox between the benefits of spatial dedication and the hazards of information overload in conventional control room alarm displays. Nuplex 80+ uses prioritization, functional organization, and digital technology to make alarm handling more manageable. Dedicated tiles now provide organizing and directing functions, but alarm information is provided through more flexible and dynamic messaging features. Lowest priority alarms are segregated from the high priority dedicated tiles. Guidance document caveats regarding multiple alarm inputs to single tiles are not applicable to the Nuplex 80+ implementation. These guidelines are concerned with the effort required to resolve the ambiguity of the alarm's source on conventional tiles. This not an issue for DIAS because it provides individual messages.

ABB-CE subsequently agreed to address alarm system concerns through additional testing using prototypes of the DIAS alarm tile display system. Item 101 was entered into ABB-CE's HF issue tracking system to provide a commitment to conduct this testing. Sub-Item g of Item 101 addresses concerns related to the use of multiple alarms and the ability of operators to access alarm information.

Note: This Table 18.8-2 was removed (See note on page 18-239)

Status: Resolved.

18.6.2.4 Findings

This review addressed:

- The methods for implementing the display and control requirements, selecting hardware and software, and refining of design concepts

- Design criteria used to determine CR and control panel arrangements including the overall configuration of the main control console and the position of individual control/display devices within individual panels

- General design characteristics that were incorporated into the HSI

These considerations were evaluated within the context of the MCR configuration, the presentation of information on controls and displays, and the layout of panels. Specific attention was given to the RCS panel and the remote shutdown panel.

This review found the application of methods, design criteria, and general design characteristics to be acceptable. Specific concerns identified included information presentation, panel layout, and configuration. ABB-CE provided responses and commitments via its HF issue tracking system to address with ABB-CE and subsequently resolved through clarification, modification, or inclusion in ABB-CE's HF TOI with a commitment to address the issue more fully at a later stage in the design process.

18.6.3.2.2 Material Reviewed

The following ABB-CE documents were referenced in this review: This is ref 11 in Section 18.1

- Reference 3 of CESSAR-DC Section 18.7, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992. ? Not CESSAR ref

- Reference 7 of CESSAR-DC Section 18.10, LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18, DSER Open Item Response," ABB-CE letter dated January 18, 1993. listed as Attached in Ref 11

- Reference 4 of CESSAR-DC Section 18.10, LD-93-135, "System 80+ Information for Issue Closure," Attachment 1, "ABB-CE Response to System 80 Operating Experience Issues Based Upon Interviews with System 80 Operators," ABB-CE letter dated September 1, 1993.

- Reference ~~3~~ 5 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 5, "SSAR-DC Markups for V&V and Procedures," ABB-CE letter dated September 24, 1993.

- Reference ~~1~~ 6 of CESSAR-DC Section 18.~~6~~4, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+," (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).

- Reference 3 of CESSAR-DC Section 18.4, "Human Factors Engineering Verification and Validation Plan for Nuplex 80+," (NPX80-IC-VP790-03, Rev. 00, September 24, 1993).

two or three depending on the context. For example, identification of symbols for active and inactive components require discrimination of the colors red and green while identification of labels for dynamic data and RG 1.97 Category 1 data requires discrimination of the colors cyan and purple. This is consistent with draft NUREG/CR-5908 Guideline 1.3.8-9, Minimum Color Differences, which states that at least 7 to 10 simultaneous colors may be discriminated if they are significantly different.

While the coding scheme was generally consistent with HF guidelines, the following issues were identified.

a. In the table of Reference 7 of CESSAR-DC Section 18.10, LD-93-005, ABB-CE identifies three alarm states (e.g., unacknowledged, acknowledged, reset) corresponding to three intensities of yellow (e.g., bright, saturated/dull, and dark). These three alarm states conflict with those provided in design description documentation - unacknowledged, acknowledged, cleared. ABB-CE was requested to clarify this discrepancy.

ABB-CE's response in Reference 14 of CESSAR-DC Section 18.10, LD-93-135 provides revised terminology for both the alarm states and the intensity levels. The four alarm states are identified as: new, existing, cleared, and reset. Reset is the null state (i.e., no alarm) and is null coded - labeled but otherwise is an empty tile outline. The three relative intensity values are identified as high, medium, and low. ABB-CE stated that the revised terminology for alarm states and intensity levels will appear in a future amendment of CESSAR-DC. Table 18.7.1-1 of CESSAR-DC is an update of the table from Reference 7 of CESSAR-DC Section 18.10, LD-93-005. The terminology in this table ~~will be~~ was modified in Amendment V to CESSAR-DC. This is part of FSER Confirmatory Item 1.1-1.

Status: Resolved.

b. The identification of alarm state requires the discrimination of three intensities of yellow. This is in conflict with Guideline 1.3.10-5 of draft NUREG/CR-5908 which states that coding by differences in brightness should be used for applications that only require discrimination between two categories of display items.

Based on a review of HF literature it was determined that the use of three intensity (brightness) levels does challenge the limits of acceptability provided by available HF guidance and, therefore, underscores the importance of testing. In addition, ABB-CE should take efforts to maximize the differences between the brightness levels. MIL-HDBK-761A states that each level of brightness coding should be separated from the next nearest level by at least a 2:1 ratio. Brightness levels selected for the alarm codes should be verified against this criteria. In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE agreed to enter this concern as Item 101 in its tracking system for open HF issues. Item 101 states that a number of concerns, including brightness coding, will be evaluated further using a prototype prior to V&V testing. ABB-CE also agreed to verify that the brightness levels in the final design vary by at least a 2:1 ratio.

Status: Resolved.

c. The coding for both active and inactive equipment status is indicated by the same switch position code (bottom) on both the CCS process controllers and the CCS switches. ABB-CE is requested to describe how the bottom switch position code is represented on the CCS process controllers and CCS switches and how the switch position code is used to identify equipment status.

In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE provided an acceptable description of the switch position codes for CCS process controllers and CCS switches. In addition, Table 18.7.1-1 of CESSAR-DC was modified to indicate that the coding scheme for two-state components is as follows: active equipment is ~~up~~ top and inactive is ~~down.~~ bottom.

Status: Resolved.

d. A color is not specified for the cross-hatch marks, which are used to indicate that a component is uncontrollable from the CCS. Should this color be considered a color code?

In Reference 14 of CESSAR-DC Section 18.10, LD-93-135, ABB-CE stated that cross-hatching is applied as a texture without color and is therefore not a color code.

Status: Resolved.

18.6.3.3.1.4 DSER Issue 18.8.2.b - System 80+ Specific Studies

DSER Issue 18.8.2, Additional HSI Information Required for Staff Review, states in part that ABB-CE should provide results of System 80+ specific studies or analyses that determine the quantitative and qualitative thresholds of "adequate" rather than "not adequate" human performance for:

3. Readability of alarm text and tiles from all operator positions in CR

5. Number of colors and shades used on displays

6. Types and amount of information encoded in the CR as well as the encoding techniques used

7. Audible and tactile feedback for controls, controllers and other devices

The number before each item corresponds to the numbers used in Section 18.6. The Items 3, 5, 6, and 7 are discussed below.

3. Readability of alarm text and tiles from all operator positions in CR.

Evaluation: ABB-CE's response cited the following sections of HFESGB as the criteria for verifying readability Section 2.5 - Equipment Labels and Section 5.3.~~1~~2 - ~~Annunciator~~ Tile Tile Matrices. The HFESGB criteria are based largely on accepted guidelines. The response did not present System 80+ specific studies. However, issues have already been entered into ABB-CE's tracking

1. Issue: System 80+ versus generic guidance - The HFESGB should provide specific guidance that have been extracted from the broad body of existing HF literature and other sources to provide rationales/justifications for specific aspects of the System 80+ design. The HFESGB should also include general guidance to support design decisions that have not yet been made.

Evaluation: A review of the HFESGB indicated that general guidance is provided when specific guidance would seem more appropriate. Requirements for specific guidance were addressed in Issues 1, 2, and 3 of Section 4.2.1. ABB-CE has provided a commitment through Item 105 of the TOI to provide this additional guidance.

Status: Resolved.

18.6.3.3.2.4 Procedure for Implementation

The designer/applicant's guideline specification document should provide an indication of how it is to be used in the overall design process.

The following are the results of a review of the HFESGB document based on these considerations.

1. Issue: Procedures that ensure systematic application of guidance to display design - While the HFESGB provides guidance regarding the details of display design, it does not provide guidance to designers to ensure systematic application of its guidelines.

Evaluation: ABB-CE provided the following commitment to apply a systematic process to display design in Appendix A, Section A-3.5.[illegible] of the HFPP:

1.6

> The reference design for the MCR and Remote Shutdown Room (RSR) indications and controls (i.e., screen design, panel layout, etc.) shall be detailed through a systematic process incorporating HFE design guidance. Appropriate documentation for the systematic process shall include the following: (1) documentation showing the results of design reviews, (2) documentation that shows how the results of the

functional task analysis are being applied to the design of specific displays, and (3) a checklist for each display page indicating important characteristics.

This response satisfactorily addresses the intent of this issue.

Status: Resolved.

2. Issue: Procedures for HSI design in non-CR environments - Neither the HFESGB nor the FTA methodology provides a procedure or guidance for systematically reviewing environmental concerns.

Evaluation: ABB-CE has committed to using task analysis as an input to the design of local control stations that are addressed by the emergency operations guidelines. Provisions have been made for recording significant environmental considerations in a miscellan~~y~~eous category. The evaluation of environmental considerations will be addressed by ABB-CE via suitability verification during verification and validation.

Status: Resolved.

3. Issue: Procedures for establishing the precision with which values are displayed - A review of the design documents and the HSI mockup indicates the apparent lack of a systematic approach for determining the degree of precision with which data are presented, whether in digital or graphic form, to the operator via the HSI.

Evaluation: Section 18.5.1.5.3 of CESSAR-DC states that parametric requirements for display and control variables will be defined in terms of device type, range, accuracy, and units of measure as part of the FTA methodology. While accuracy of data is an important requirement, it is a separate concern from the precision with which data is presented to the operator (e.g., the number of significant digits in digital displays, the number of intervals on scale displays). Specific examples of the lack of clearly defined display precision requirements were observed during the design features review with respect to the scaling on bar charts and other indicators. The HFESGB provides general criteria for scaling. However, this is insufficient for determining the precision requirements for specific parameters.

In Amendment S to the CESSAR-DC, ABB-CE modified Section 18.5.1.5.3 to require that parametric requirements for display and control variables be defined in terms of precision, in addition to, device type, range, accuracy, and units of measure. This entry will state that the display precision of each measured variable is provided based on operator task requirements. In addition, ABB-CE has modified Section 6.1.5.2 - Phase 2 Availability Inspection Criteria of its V&V plan (NPX80-IC-VP790-03) to indicate that precision specifications will be verified for each as-built item of the HSI. These modifications were found to satisfy the intent of the review issue.

Status: Resolved.

18.6.3.3.3 Specific Issues

This section provides a review of issues and concerns that were identified through reviews of selected guidelines of the HFESGB or were identified during the evaluation of other review issues.

1. Issue: Symbols and graphical formats - Graphic forms (e.g., bar charts and deviation bar charts) that are used in the HSI displays and controllers are not adequately described in the HFESGB.

Evaluation: The symbols and graphical formats included in HFESGB were found to be generally consistent with those used within the nuclear power industry. The specific implementation of these symbols and graphical formats will be evaluated during verification and validation when the design is complete. However, the review indicated that the specific graphic forms (e.g., bar charts and deviation bar charts) that are used in the HSI displays and controllers are not adequately described in the HFESGB.

ABB-CE agreed to provide in a future revision of the HFESGB specific additional guidance pertaining to graphic formats used in DPS displays such as bar different from that used in the prototype. ABB-CE was requested to consider specifically evaluating text legibility under conditions that are representative of actual use. ABB-CE agreed and entered issue 102 into the TOI to ensure that the legibility of controls and displays will be evaluated under conditions that are representative of anticipated work conditions.

Status: Resolved.

3. Issue: The criteria for minimum size of poke areas for touch screens specified in HFESGB is inconsistent with draft NUREG/CR-5908 and other guidelines.

Evaluation: Guideline 3.4.9.1 of HFESGB specifies that touch target areas should have a minimum height of 6 mm (0.25 in.), a minimum area of 161 mm^2 (0.25 sq in.) and a resulting minimum width of 25 mm (1.0 in.). In addition, the HFESGB criteria for separation of touch target areas is unclear. Guideline 3.2.4-10 of draft NUREG/CR-5908 specifies a minimum height and width of 15 mm (0.6 in.) with a resulting minimum area of 232 mm2 (0.36 sq in.). Therefore, the size recommended by draft NUREG/CR-5908 is over 40 percent larger than the area specified by HFESGB. ABB-CE was requested to address this apparent inconsistency.

ABB-CE's basis for Guideline 3.4.9.1 of HFESGB was subsequently reviewed in greater depth including consideration of unique characteristics of the System 80+ touch screens such as the provision of visual feedback when the touch area is entered and the "make on break" mode of actuation. Based on these considerations the minimum touch area dimensions were found to be acceptable. In addition, ABB-CE agreed to clarify the wording of the criteria for separation of touch target areas.

Status: Resolved.

4. Issue: Anthropometric dimensions.

Evaluation: The bases for HFESGB Guideline 7.5.2.1 provides a discussion of anthropometric data pertaining to the distance from the central axis of the body to the panel edge and the eye distance forward of the central axis of the body. However, the use of these dimensions in the evaluation of panel dimensions is not clear in this discussion.

ABB-CE's response in Reference 2 provided an adequate explanation for the discussion of these dimensions.

Status: Resolved.

5. Issue: Justification of vision and reach envelops on System 80+ control panels.

Evaluation: The description of control panel dimensions found in Section 18.6.5.7 of CESSAR-DC states that the anthropometric data for these profiles are based on the HFESGB and MIL-STD-1472D. Further justification for the specific dimensions of these panels was not clear based on the material presented in Section 7.6.2.1 of HFESGB. This section states that the reach envelopes are unique to each panel according to bench board depth and slope and must be evaluated individually. Based on this review the following concerns were identified.

a. Figures 18.6.5-11 and 18.6.5-12 of CESSAR-DC show eye heights for a 95th percentile man and a 5th percentile woman. This position is aligned with the leading edge of the bench board. Why are not these positions set off horizontally to allow for torso and head width? What are the resulting maximum viewing distances for the man and woman? How do they compare to the specified viewing distances for controls and displays?

ABB-CE's response in Reference 2 states:

Reference source not identified

The "at-the-panel" viewing distance in HFESGB Section 2.2.3.2 (i.e., 91 cm (36 in.)) is only slightly greater than the 95th percentile male reach envelope, which is shown to easily capture the panel work surfaces. Maximum viewing distances imposed by the panels, which are most limiting for the 5th percentile female, are less than 76 cm (30 in.) for standup panels, less than 79 cm (31 in.) for sitdown

* Significant human actions identified in the HRA/PRA

* Initiating events to be considered in the EOPs shall include those events present in the design bases

Evaluation: The information addressed by DSER Issue 18.9.1 generally satisfy this criterion as well.

Status: Resolved.

18.7.4 Findings

The staff found that the ABB-CE approach to System 80+ procedure development was acceptable and that the information that will be provided by ABB-CE to the COL applicant is generally satisfactory to support the development of plant operating procedures.

18.8 <u>Verification and Validation</u>

The NRC HFE PRM for advanced evolutionary reactors specified that a formal V&V (Element 8) of the HSI should be performed. The staff's DSER review of the CESSAR-DC has identified a DSER issue related to HFE PRM Element 8 (i.e., DSER Issue 18.10-1).

18.8.1 Objectives

The objective of this review is to provide comments on the ABB-CE plan related to HFE PRM Element 8 - Verification and Validation.

18.8.2 Methodology

18.8.2.1 Material Reviewed

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

- Reference ~~15~~ 16 of CESSAR-DC Section 18.10, LD-93-071, "System 80+ Submittal #1 Design Descriptions and ITAAC," ABB-CE letter dated April 30, 1993.

- Reference ~~16~~ 17 of CESSAR-DC Section 18.10, LD-93-140, "System 80+ Information for Issue Closure," Attachment 2, "Justifications of ABB Positions Requested for Closure of V&V;" and Attachment 5, "SSAR-DC Markups for V&V and Procedures;" ABB-CE letter dated September 24, 1993.

- Reference 3 of CESSAR-DC Section 18.7, LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), ABB-CE letter dated May 8, 1992.

- Reference 3 of CESSAR-DC Section 18.4, "Human Factors Engineering Verification and Validation Plan for Nuplex 80+" (NPX80-IC-VP790-03, Rev. 00 September 24, 1993), hereafter referred to as the plan.

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

- CESSAR-DC, Sections 13.5 and 18.9.

18.8.2.2 Review Scope

The scope of this review was centered on the V&V plan, although additional ABB-CE documents were consulted (as referenced above).

The review focused on (1) resolution of DSER issues, and (2) evaluation of the ABB-CE documents with respect to the topics and general criteria of the HFE PRM. Complete adherence to the HFE PRM was not considered to be mandatory. Differences in approach would be considered acceptable provided (1) the program can still meet the HFE commitment and goals, (2) the difference between the proposed criteria and those contained in the HFE PRM are adequately justified, and (3) there is no adverse impact on other program elements.

is it clear how they were used. For example, EPRI NP-3701, Computer-Generated Display System Guidelines, was referenced; yet in the section on suitability verification (where the document would most likely be applied), it was not identified as a criteria document. An examination of the verification analysis report did not indicate that anything other than NUREG-0700 was used for verification (which does not contain sufficient criteria verification of a CR such as the System 80+). The revision to the plan provided the specific references to the HFE PRM recommended technical basis documents. The verification analysis will be based upon criteria from a broad basis of HFE PRM identified documents (including, but not limited to, NUREG-0700) and additional acceptable industry sources. Four HFE PRM identified sources were noted (AR 602-1, TOP 1-2-610, DODI 5000.2, and EPRI NP-3701). These exceptions were acceptable to the staff since their contribution to the ABB-CE V&V effort was redundant with the documents cited. (The review and acceptance of the specific criteria used for verification is addressed in FSER Section 18.6.3, HFE Standards, Guidelines, and Bases.) The staff, therefore, determined that the technical basis of the ABB-CE V&V plan was acceptable. Based upon the revisions to the plan, this issue is resolved.

Status: Resolved.

18.8.3.2.3 Human Factors Issue Resolution Verification

Criterion: HFE PRM Criterion 7 states that a verification shall be made that all issues documented in the HF issue tracking system have been addressed.

Evaluation: The staff noted that verification of HFE issues resolution was not addressed in the draft plan. Following discussions with the staff, ABB-CE addressed the concern in two ways. First, assurance of closeout of tracking system items is incorporated into the description of the tracking of issues description of the HFPP. Second, plan Sections 6.1.4 (Availability Verification), 6.2.~~2~~4 (Suitability Verification), and 6.3.4.1 (Validation) were modified to require that relevant TOI items are addressed in the appropriate V&V activity. Since final closeout of the TOI items is a COL activity, the consideration is how CR personnel will use paper procedures in the CR. This includes considerations of task lighting, ease of handling, and adequacy of laydown. Similar evaluations should consider P&IDs, TSs, and other operator aids.

d. Operator awareness of the status of equipment under surveillance test or repair.

2. HSI Design: It is the staff's position that the evaluation of the DPS and DIAS alarm implementation under high-alarm conditions should be specifically evaluated in validation.

Evaluation: These issues were not addressed in the draft plan. Following discussions with the staff, ABB-CE has addressed the issues in the following ways. With respect to Item 1a and b, plan Section 6.3.4.2 has been revised to include basic maintenance tasks during normal operations. With respect to Item 1c, the issues will be addressed during suitability analysis and any concerns observed during validation testing will be noted. With respect to Items 1d and 2, requirements for their evaluation ~~have been~~ will be included as part of the implementation plan appendix (plan Appendix B).

Based upon this plan revision, these issues are resolved.

Status: Resolved.

18.8.3.2.7 Scheduling

Criterion: In the proposed resolution of the DSER, ABB-CE agreed to provide a schedule of V&V activities (as per the HFE PRM requirement in Element 1).

Evaluation: Scheduling is described in plan Section 7. In Section 7.1 it states that availability verification can be accomplished "in parallel with, before, or after suitability verification." The staff was concerned that not all HSI changes resulting from availability verification would be subject to suitability verification. Following discussions with the staff, ABB-CE modified Section ~~7.2~~ 6.2.2 of the plan to indicate that all HSI items will be verified as suitable. Based upon this plan revision the issue is resolved.

Status: Resolved.

18.8.4 Findings

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

18.9 <u>Certified Design Description/Inspections, Tests, Analyses, and Acceptance Criteria</u>

18.9.1 Objectives

The objective of this review is to evaluate the System 80+ MCR ITAAC, remote shutdown room ITAAC, and control panels ITAAC against the requirements of 10 CFR Part 52.47(a)(1)(vi) and the HFE PRM.

18.9.2 Methodology

18.9.2.1 Material Reviewed

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

- System 80+ ITAAC Section 12.2.1, "Main Control Room;" Section 2.12.2, "Remote Shutdown Room;" and Section 2.12.3, "Control Panels."
- Reference 1~~5~~6 of CESSAR-DC Section 18.10, LD-93-071, "System 80+ Submittal #1 Design Descriptions and ITAAC," ABB-CE letter dated April 30, 1993.

- Reference 13 of CESSAR-DC Section 18.10, LD-93-147, "System 80+ Information for Issue Closure," Attachment 1, "Response to Cross-Branch Chapter 18 Questions (October 4, 1991)," ABB-CE letter dated October 18, 1993.

18.9.2.2 Review Scope

The scope of this review was centered on the following System 80+ ITAAC and associated design descriptions: ITAAC Number 2.12.1, "Main Control Room;" ITAAC Number 2.12.2, "Remote Shutdown Room;" and ITAAC Number 2.12.3, "Control Panels."

The review focused on ensuring that significant features of the design certification application contained in the CESSAR-DC are captured by the CDD.

18.9.2.3 Review Procedure

As indicated above, the staff's DSER review of the CESSAR-DC indicated that ABB-CE must provide appropriate CR HF ITAAC, including DAC for portions of the design not completed at the time of the final design approval. Further, the staff noted that the ITAAC and DAC should be consistent with the criteria described in the HFE PRM.

By letter dated April 30, 1993 (Ref. 1~~5~~6 of CESSAR-DC Section 18.10, LD-93-071), ABB-CE submitted to the NRC for review and approval ITAAC and associated design descriptions for the MCR, remote shutdown room, and control panels.

The ITAAC and CDD were reviewed using the requirements of the HFE PRM and Part 52. Staff comments were discussed with ABB-CE at the public meeting held October 4 through 6, 1993. The resolution of staff comments were documented in minutes of that meeting.

The following materials were consulted as part of the evaluation:

- NRC HFE Program Review Model for Evolutionary Reactors (HFE PRM).

design commitment in the MCR and RSR ITAACs (i.e., design commitment 4). In addition, task analysis output is an input to the availability inspection acceptance criteria of design commitment 2, as discussed in the System 80+ V&V plan. The design description statement in the control panels ITAAC regarding control panel seismic category ~~will be~~ was added to the MCR and RSR design descriptions. The staff found this approach and commitments acceptable.

18.9.3.2 Level of Detail

The ITAAC were evaluated to assure that they accurately reflected the design and implementation process and that they were at a level of detail consistent with the staff's intent to not constrain the use of state-of-the-art, proven technology at the time the HSI is designed (one of the stated intents of the DAC concept). All necessary and sufficient ITAAC were identified based upon comparison to the HFE PRM and no concerns were identified.

Therefore, the staff concludes that the design commitments in the HFE ITAAC/DAC accurately summarize the Design Description for HFE; that the inspections, tests, and analyses identified are acceptable methods for determining whether the design commitments have been met; and that the acceptance criteria are sufficient to establish, if they are met, that the design commitments have been met.

18.9.3.3 Main Control Room Minimum Inventory

18.9.3.3.1 Discussion in the CESSAR-DC

ABB-CE's initial CESSAR-DC provided insufficient information about controls, displays and annunciators to be utilized for the System 80+ CR, resulting in a staff RAI. As part of the general resolution of the lack of CR detail, ABB-CE provided the detailed CR design implementation process, through which the specific controls, displays, and annunciators will be specified and designed. However, in order to provide an initial set of controls, displays, and annunciators for transient mitigation before design certification, ABB-CE developed the inventory described in CESSAR-DC Chapter 18. This inventory was developed by analyzing the System 80+ EPGs and the important operator actions

References

ASEA-Brown Boveri Combustion Engineering (1993). "System 80+ Standard Safety Analysis Report-Design Certification" (through Amendment S). Windsor, CT.

International Electrotechnical Commission (1989). "International Standard: Design for Control Rooms of Nuclear Power Plants" (IEC-964). Geneva, Switzerland: Bureau Central de la Commission Electrotechnique Internationale.

Kockler, R., Withers, T., Podiack, J., and Gierman, M. (1990). Systems engineering management guide (Department of Defense AD/A223 168). Fort Belvoir, VA: Defense Systems Management College.

?

LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "System 80+ Function and Task Analysis Final Report" (January 1989), the applicant letter dated May 8, 1992.

LD-92-065, "System 80+ Supplements to RAI Responses," Attachment 4, "Nuplex 80+ Verification Analysis Report" (NPX80-TE790-01, Rev. 02, December 1989), the applicant letter dated May 8, 1992.

LD-92-076, "System 80+ Shutdown Risk Report, Revision 1," Attached, "System 80+ Shutdown Risk Evaluation Report" (DCTR, June 15, 1992), the applicant letter dated June 16, 1992.

LD-92-102, "System 80+ Human Factors Documentation Submittal," Attathment 1, "Nuplex 80+ Advanced Control Complex Design Bases" (NPX80-IC-DP-790-01, Rev. 00, January 15, 1990), the applicant letter dated September 23, 1992.

LD-92-102, "System 80+ Human Factors Documentation Submittal," Attachment 2, "Nuplex 80+ Compliance With NUREG-0737 Supplement 1 Requirements," the applicant letter dated September 23, 1993.

LD-92-120, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment (untitled), Response to DSER Issue Nos. 20.2-23 and 20.2-29, the applicant letter dated December 18, 1992.

LD-93-005, "Closure of System 80+ Draft Safety Evaluation Report Issues," Attachment 5, "Chapter 18, DSER Open Item Response," the applicant letter dated January 18, 1993.

LD-93-071, "System 80+ Submittal of Design Descriptions and ITAAC," the applicant letter dated April 30, 1993.

LD-93-135, "System 80+ Information for Issue Closure," Attachment 1, "ABB-CE Response to System 80 Operating Experience Issues Based Upon Interviews with System 80 Operators," the applicant letter dated September 1, 1993.

LD-93-140, "System 80+ Information for Issue Closure," Attachment 2, "Justifications of ABB Positions Requested for Closure of V&V;" and Attachment 5, "SSAR-DC Markups for V&V and Procedures;" the applicant letter dated September 24, 1993.

LD-93-147, "System 80+ Information for Issue Closure," Attachment 1, "Response to Cross-Branch Chapter 18 Questions (10/4/91)," the applicant letter dated October 18, 1993.

Nuclear Management and Resources Council (1991). "Guidelines for Industry Actions to Assess Shutdown Management" (NUMARC 92-106). Washington, DC.

Nuclear Safety Analysis Center (1981). "Verification and Validation for Safety Parameter Display Systems" (NSAC-39). Palo Alto, CA.

NUPLEX 80+ Verification Analysis Report (NPX80-TE790-01) Revision 2, 12/13/89.

Reference ~~1~~ 6 of CESSAR-DC Section 18.~~6~~4, "Human Factors Engineering Standards, Guidelines, and Bases for System 80+," (NPX80-IC-DR-791-02, Rev. 00, September 15, 1993).

2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plants. The Commission recognizes the importance of mitigating the consequences of a core-melt accident and continues to emphasize features such as containment, siting ~~sighting~~ in less populated areas, and emergency planning as integral parts of the defense-in-depth concept associated with its accident prevention and mitigation philosophy. The Commission approves use of the qualitative safety goals, including use of the quantitative health effects objectives in the regulatory decision-making process.

19.0.1.3 Standardization Policy Statement

The Commission issued the Policy Statement on Nuclear Power Plant Standardization on September 15, 1987. The policy statement encouraged the use of standard plant designs and provided information concerning the certification of plant designs that are essentially complete in scope and level of detail. The intent of these actions was to improve the licensing process and to reduce the complexity and uncertainty in the regulatory process for standardized plants. In relation to severe accidents, the policy statement expected applicants for a design certification to address the four licensing criteria for new plant designs set forth in the Commission's Severe Accident Policy Statement.

19.0.1.4 10 CFR Part 52

The Commission issued 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," on April 18, 1989. This rule provides for issuance of early site permits, standard design certifications, and combined licenses with conditions for nuclear power reactors. It sets out the review procedures and licensing requirements for applications for these new licenses and certifications and

FSER Section 19.1.2.1

Page 19-30, Separate Startup and EFWS/Four Train EFWS

Delete text refering to automatic initiation of the Startup Feedwater System (see markup of page 19-30). The Startup Feedwater System design was changed from automatic initiation to manual initiation after the level 1 PRA was completed. This design change was identified in Table 19.6A-1 as a difference between the PRA models and the current design. This design change is not expected to have a significant impact on the core damage frequency.

Safety Depressurization System (SDS)

An important function of the SDS is the manual safety-grade means of rapidly depressurizing the RCS so that SI can be actuated, when DHR fails via either the SGs or the SCS, to perform core cooling by "feed and bleed" operation. The rapid depressurization function of the SDS constitutes the "bleed" portion of the "feed and bleed" operation while SI constitutes the "feed" portion. This is an important feature added to the System 80+ design that aims to reduce the failure probability of long-term DHR. SDS has also a mitigative function. Rapid depressurization can be used to mitigate some of the potential containment challenges associated with reactor vessel failure at high pressure (see Section 19.1.2.2 below).

Multiple Independent Connections to the Grid and Turbine/Generator runback capability

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine generator system and the associated buses are designed to run back to maintain hotel loads on a loss of grid. These features aim at reducing the frequency of LOOP initiating events and therefore the frequency of accident sequences that are associated with LOOP including SBO.

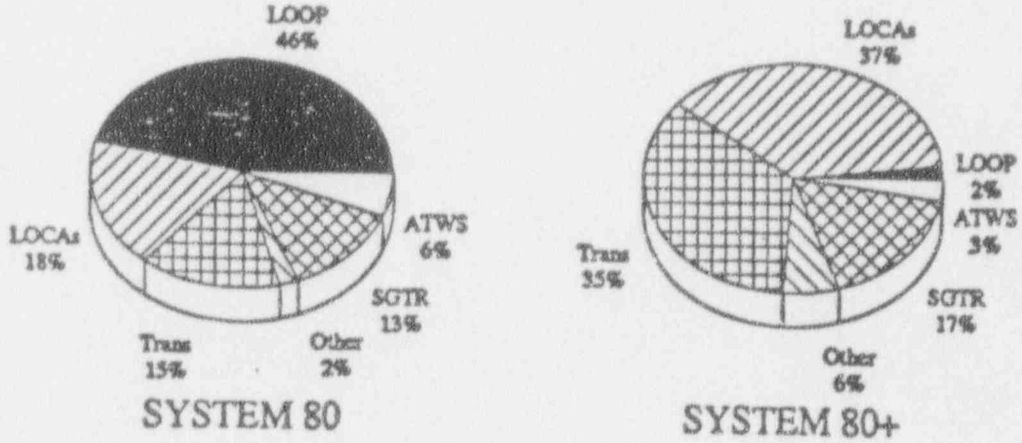
Separate Startup and EFWS/Four Train EFWS

The use of a non-safety grade startup feedwater system (SFWS) for normal startup and shutdown operations helps reduce the demands on the EFWS. In addition, the SFWS provides an independent means of supplying feedwater to the SGs for removing heat from the RCS during emergency conditions when the main feedwater is not available ~~(it is automatically actuated upon loss of main feedwater before the EFWS)~~. The EFWS is a dedicated system, which provides an independent safety-related means of supplying feedwater to the steam generators for the early phase of DHR in the event that ~~both~~ normal feedwater ~~and startup feedwater are~~ is lost. The EFW system consists of two trains, each train aligned to feed its respective steam generator. Each train contains one

Table 19.1.1. Comparison of Core Damage Frequency Contributions by Initiating Event

Initiating Event	System 80 (CDF/yr)	System 80+ (CDF/yr)
Large LOCA	2E-6	1E-7
Medium LOCA	4E-6	3E-7
Small LOCA	1E-5	2E-7
Steamline/Secondary Line Break (SLB)	1E-6	2E-9
Steam Generator Tube Rupture (SGTR)	1E-5	3E-7
Transients	1E-5	6E-7
Loss of Offsite Power (LOOP)	4E-5	~~4E-8~~ 3E-8
Anticipated Transient Without Scram (ATWS)	5E-6	5E-8
Interfacing System LOCA	5E-9	5E-10
Vessel Rupture	1E-7	1E-7
Total	8E-5	2E-6

Figure 19.1.1. Relative Contributions to Total CDF From Internal Events.



- Start-up feedwater system, with source from the CST ~~and actuated before the EFWS~~ -- contributes to the increased reliability of heat removal through the SGs.

- Turbine-generator full run-back capability -- reduces initiating event frequency.

- Two redundant and diverse EFW actuation systems -- increases the reliability of secondary heat removal.

The following are the most important features of the System 80+ design which contribute to the reduction in the estimated CDF associated with steam generator tube rupture (SGTR) sequences (CDF reduced from 1E-5 per year to 3E-7 per year):

- Four train EFWS -- the increased reliability of this system (four instead of two or three trains) reduces the reliance on "feed and bleed" cooling as the last defense against core damage (for System 80+ the RDS can be used for "feed and bleed" cooling).

- Four train SIS -- the increased reliability of this system (four instead of two trains) reduces the importance of performing "aggressive secondary cooldown (ASC)" for early core cooling. ASC, which is the last line of defense when SI is not available, requires use of both SGs and involves rather complicated human actions to be performed in short times.

- SDS -- provides an alternate DHR path through primary "feed and bleed" which is much more reliable and faster than the high pressure "feed and bleed" cooling of currently operating PWRs (replacing PORVs by MOVs simplifies operator actions and provides flexibility for controlled and fast depressurization to SIS actuation pressures).

- Large IRWST capacity with refill capability -- increases the long-term recovery probability for unisolable SG leaks, which bypass containment, by preventing depletion of borated water and core damage.

- Operator failure to perform the following actions were found to be major contributors to the estimated CDF from internal events; (i.e., these actions have the highest "risk reduction worth"):

  - perform "aggressive secondary cooldown"

  - initiate "feed and bleed" operation

As mentioned above, details on SSCs and human actions that were found to be risk significant by ABB-CE are documented in CESSAR-DC Section 19.15. This information, which was generated by taking into account insights and assumptions from the entire PRA (i.e., all three PRA levels for both internal and external events and for all modes of operation), form the basis for the following two lists: (1) a list of important SSCs (see Table 19.15.6-1) which the COL applicant should be incorporated in the D-RAP and O-RAP programs (COL Action Item 19.14); and (2) a list of risk important ("critical") operator tasks (see Table 19.15.6-2) which should be taken into account in the MCR V+V ~~design~~ process, as well as in the development of emergency procedures and training programs.

ABB-CE, in performing the level 1 PRA for internal events at power operation, identified the following 10 "critical" tasks, which must be performed by the operator to prevent or mitigate severe accidents, that should be taken into account in the MCR design and the fixed display panel. ABB-CE provides a commitment to do this in Section 18.5.1.5.2 of the SSAR (Amendment Q). The process for inclusion of these tasks and the acceptability of this approach is addressed in Section 18.5.3.2.2 of this report.

- Operator fails to initiate hot leg injection (HHFFHOTLEG).

- Operator fails to align the CST to EFWSTs (AHFDCST).

- Operator fails to initiate "feed and bleed" (VHFFFEEDBLEED).

- Operator fails to align the SCS for injection operation (JHFDRHRI).

the CCFP would be about 3 percent. The CCFP based on the dose definition is lower than that associated with the structural integrity definition since the bulk of the structural failures (such as basemat melt-through failures) do not have significant offsite consequences.

The staff concludes that the estimated CCFP for the System 80+ design satisfies the Commission's containment performance goal (0.10). Specifically, within the 24 hour period following core damage, which is the focus of the containment performance goal, the probability of containment failure (using either the structural integrity or dose definition of containment failure) is below the goal. The probability of containment failure is somewhat higher when failures beyond 24 hours are included, however, CCFP remains less than the goal using the dose definition of failure (3 percent), and is only slightly higher than the goal (11 percent) using the structural definition of failure. In SECY-90-016, the staff stated that in view of the low probability of accidents that would challenge the integrity of the containment, the CCFP for evolutionary designs should not exceed "approximately" 0.1. Furthermore, in the related SRM the Commission directed that the CCFP objective of 0.1 should not be imposed as a requirement in and of itself. In view of the approximate nature of the containment performance goal, the recognition that PRA results, particularly bottom line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11 percent CCFP estimate are late, containment basemat melt-throughs rather than releases to the atmosphere, the staff concludes that the System 80+ design satisfies the Commission's containment performance goal. ~~even given this deviation~~.

19.1.3.2.2 Leading Contributors to Containment Failure from the Level 2 PRA

The frequencies of the various containment failure modes and the fractional contributions by containment failure mode to the total containment failure frequency are presented in Figure 19.1.2 for both the original and updated System 80+ PRA. The updated PRA results reported here, as well as in the discussions that follow, are based on the combined frequency of internally-initiated events plus tornado strike events. A separate accounting of results for internally-initiated events is not included due to the small CDF from

This design commitment is consistent with results of NRC funded research into Seal and Penetration integrity during Severe accidents.

be achieved by a combination of selecting high quality and high capability seals, protectively mounting the seal so that it is not directly exposed to the containment environment, and providing double seals (inner and outer) whenever possible. This design commitment is provided in CESSAR-DC Section 19.11.3.1.4. The staff considers ABB-CE's design objective and the assumed failure rate in the PRA achievable ~~provided the containment penetration seal materials are procured to maintain stability at temperatures up to 500 °F~~. This is discussed further in Section 19.2.6.4 of this report.

Early Containment Failure

The total frequency of early containment failure predicted by the System 80+ PRA is 2E-8 per year, or about 11 percent of the containment failure frequency. The mechanisms that are considered in the System 80+ PRA for early containment failure include in-vessel steam explosion (alpha mode failure), direct containment heating (DCH), early hydrogen burn, rapid steam generation (RSG), rocket mode failure, and corium impingement due to high pressure melt ejection.

The major contributors to early containment failure are those from energetic events such as steam explosion and hydrogen detonation. The fractional contributions from the various containment failure mechanisms to early containment failure are 88 percent for ex-vessel fuel coolant interactions, 9 percent for alpha mode failure, and 3 percent for early hydrogen burn. The contributions from DCH, rocket mode failure, and corium impingement are very small (less than 0.01 percent).

Ex-vessel fuel coolant interactions are the leading contributor to early containment failure according to the System 80+ PRA. These events include ex-vessel steam explosion and quasi-static pressurization (i.e., a pressure spike produced in the containment by the steam generated from the quenching of the high temperature core debris). According to the System 80+ PRA, ex-vessel steam explosions are about an order of magnitude more likely to cause a containment failure than quasi-static pressurization. The high contribution of steam explosions to early containment failure can be attributed to the high probability that the reactor cavity will be flooded using the CFS (over "scoping" fire risk analysis performed by ABB-CE compliment this belief. The fire risk analysis has provided useful safety insights for inclusion in ITAAC, COL action items, and RAP. Since detailed PRA-based internal fire analyses at some operating plants have shown that fire-induced sequences can be leading contributors to CDF, ABB-CE has stated (COL Action Items 19.5, 19.7, and 19.12) that the COL applicant should provide an updated internal fire PRA that takes into account design details, such as cable routing and door locations as well as fire detection and suppression system location, to search for internal fire vulnerabilities in the detailed design.

The staff's review determined that there are 11 doors above the 70+ level in the System 80+ design where the door penetrates the dividing divisional wall (i.e., hard concrete interdivisional barrier) between the two divisions. ABB-CE provided additional details on the System 80+ design to justify that these 11 doors do not constitute potential fire vulnerabilities. ABB-CE indicated that all of these doors are self-closing and are alarmed in the MCR. In addition, there are always additional intervening fire doors (self-closing doors and alarms) between the doors in the divisional wall and any safety related equipment or equipment credited in the System 80+ PRA.

RCP seal LOCAs are not modeled as a credible event in the internal events analysis. Instead ABB-CE provided a sensitivity study for at-power events on the potential effects of loss of cooling to the seals. However, due to the staff's concern about the potentially much higher common mode failure rate associated with fire events, ABB-CE provided an evaluation of loss of seal cooling due to a fire and resultant RCP seal LOCA. Based on this submittal, the staff finds that RCP seal LOCAs do not constitute a vulnerability during severe accident fires. However, the staff believes that the available test results for the RCP seals do not provide full confidence in the capabilities of the seals under loss of seal cooling conditions. Therefore, for the purposes of defense in depth, the dedicated seal injection pump (an air-cooled positive displacement pump) should be located in such a manner as to minimize its vulnerability to internal fires and floods. that could also affect the primary means of providing RCP seal cooling or RCP seal injection,

The System 80+ design has significant robustness to prevent and mitigate severe accident fires and the design should result in a plant with superior results for the RCP seals do not provide full confidence in the capabilities of the seals under loss of seal cooling conditions. Therefore, for the purposes of defense in depth, the diverse positive displacement RCP seal injection pump (air-cooled) should be located in such a manner as to minimize its vulnerability to internal fires and floods that could also affect the primary means of providing RCP seal cooling or RCP seal injection.

The System 80+ design has significant robustness to prevent and mitigate severe accident floods and the design should result in a plant with superior capabilities to prevent and mitigate floods compared to operating nuclear power plants.

19.1.4.3.1 Dominant Accident Sequences (Internal Floods)

ABB-CE's "scoping" flood risk analysis for the System 80+ design used applicable event and fault tree models from the internal events analysis (equipment needed to mitigate "transient" events) to identify the following dominant accident sequences:

- Flood disables one division of ESF, there is failure of the DHR system, and failure of SDS valves to reduce pressure to perform feed and bleed.

- Flood disables one division of ESF, there is failure of the DHR system, and the SI pump fails to provide feed for feed and bleed.

- Flood disables one division of ESF, there is failure of all feedwater systems, and failure of safety depressurization valves to reduce pressure so can perform feed and bleed.

- Flood disables one division of ESF, there is failure of all feedwater systems, and the SI fails to feed for feed and bleed.

19.1.4.3.2 Risk Important Design Features & Human Actions (Internal Floods)

The following is a list of some of the design features that are responsible for reducing the impact of an internal flood in the System 80+:

requiring that two of the four SIS pumps be available in shutdown modes when the IRWST is available.

- Design features which are important for preventing and mitigating LOOP/SBO events during power operation, are also important in reducing the frequency of these events during shutdown operation. These are: (1) two separate and independent switchyards, (2) redundant and diverse onsite AAC power sources (two EDGs and a CTG). The following operational requirements are important during shutdown operation:

  - The reliability of the two switchyards is an important feature contributing to the reduced System 80+ shutdown risk from LOOP/SBO events, as compared to operating reactor designs. When a switchyard is unavailable for maintenance, the COL applicant should ensure that no activities which could fail the operating switchyard are taking place and no fire sources are present. This is COL Action Item 19.17. transient

  - The reliability of the redundant and diverse emergency onsite ac power sources is an important feature contributing to the reduced shutdown risk from LOOP/SBO events, as compared to operating reactor designs. For this reason, a new TS was added requiring that two of the three onsite emergency ac power sources (i.e., two EDGs and a CTG) be available during shutdown operation.

- Due to the increased reliance on human actions and greater opportunity for human errors during plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures, training, and operator knowledge of plant configuration. The control of these activities is an important COL applicant responsibility. This is COL Action Items 19.15 and 19.17.

- During plant shutdown, the integrity of fire and flood barriers between areas in same division, such as quadrants, where systems comprising the alternate shutdown success paths are located, should be maintained. This will require configuration control of fire/flood barriers for determine if any vulnerabilities have been introduced. This is COL Action Item 19.9.

The COL applicant should incorporate the information on risk important operator tasks (see list of Table 19.15.6-2) in the MCR ~~design~~ V and V Process. The COL applicant should also use this information in developing and implementing procedures, training and other human reliability related programs. This is COL Action Item 19-15 The COL applicant is also responsible for developing detailed procedures for actuation and operation of the severe accident design features, such that the use of these features is consistent with the PRA assumptions, or for modifying the site-specific PRA to match the modified assumptions. This is COL Action Item 19-16.

Integrity of divisional separation between redundant safety-related equipment is a key assumption in the System 80+ fire and flood risk analyses. This divisional separation, which is extended also in the SSW/CCW structures, prevents fires and floods from propagating from one division to the other. There are no doors or passageways connecting the divisions of safety-related equipment up to elevation 70+0.

Separate ventilation systems for each division minimizes the possibility of smoke, hot gases, and fire suppressants migrating from one division to another.

Electrical separation between the two safety-related divisions is maintained.

All drains are divisionally separated. Drains within a division, drain to the lowest level which has adequate volume to collect water from a break in any division. They are sized to handle the potential discharge of fixed fire suppression systems and fire hoses.

During plant shutdown operation, the integrity of fire and flood barriers between areas in same division, such as quadrants, where systems comprising the alternate shutdown success paths are located, should be maintained. This will require configuration control of fire/flood barriers for shutdown operation by the COL applicant. This is COL Action Item 19.18. The COL

Drains in the nuclear annex and the reactor building are divisionally separated, have seismic Category I valves to prevent backflow, and are sized to handle fire protection system discharges. Each subsphere quadrant has its own redundant seismic Category I sump pumps that can be powered off the EDGs.

The possible sources of internal flooding within the nuclear annex and reactor building are located below elevation 70+0.

The seals for the underground pipe chase (contains CCW piping) between the nuclear annex and the CCW building will be capable of withstanding an internal flood from a pipe break in the CCWS/SSWS building (e.g., service water).

Chemical and Volume Control System (CVCS)

Divisional separation exists between redundant charging pumps and their power supplies.

There will be diverse RCP seal injection capability using positive displacement pump that is diverse from the CVCS and can be powered from either the EDGs or the CTG. The dedicated seal injection pump (air-cooled positive displacement pump) should be located in such a manner as to minimize its vulnerability to internal floods and fires that could also affect the primary means of RCP seal cooling or RCP seal injection.

Instrument Air System (IAS)

Divisional separation exists between redundant trains of instrument air and their power supplies.

Instrumentation and Control

To provide sufficient diversity and defense in depth to mitigate all postulated accidents even assuming a common cause failure within the plant protection system, the System 80+ instrumentation and control systems provide the manual hardwired ESFAS for the controls and for display there are hardwired key indications of critical function status for post accident monitoring.

depressurization capability of the SDS, the IRWST, and the reactor CFS) which contributed to the reduced CDF and CCFP estimates of the System 80+ design when compared with operating PWRs. PRA results and insights were used to identify areas where it is particularly important to implement the design and operational requirements assumed for design certification (e.g., ITAACs, D-RAP, O-RAP, TSs, operator training and procedures). Based on this review the NRC believes that the System 80+ design represents an improvement in safety over operating PWRs in the United States.

References

1. SECY Paper Memorandum, Samuel J. Chilk (NRC) to James M. Taylor (NRC), "SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," July 21, 1993

19.2 Severe Accident Performance

19.2.1 Introduction

The purpose of Section 19.2 is; (1) to consolidate the NRC's approach to resolution of severe accident issues for advanced light water reactors as specified in SECY-90-016, SECY-93-087, and the corresponding SRMs, and (2) to evaluate the approach proposed by ABB-CE for resolution of severe accident issues for the System 80+.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense in depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. ~~Sighting~~ Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population.

The reactor and containment design provide a vital link in the defense-in-depth philosophy. Current reactors and containments are designed to withstand

19.2.2 Deterministic Assessment Of Severe Accident Prevention

19.2.2.1 Severe Accident Preventative Features

The System 80+ is designed to cope with plant transients and loss of coolant accidents without any adverse impact on the environment. However, the potential does exist, albeit remote, for a LOCA or seemingly ordinary plant transient coupled with numerous plant safety system failures to progress to a severe accident with the potential for substantial offsite releases.

Accident initiators can be separated into two general groups - transients and LOCAs. Transients include planned reactor shutdowns and transients which result in reactor scrams. Examples of transients include: manual shutdown, steamline break (SLB), steam generator tube rupture (SGTR), LOOP, loss of feedwater. In addition to these transients, there is an entire spectrum of LOCAs which are accident initiators. LOCAs generally fall within three categories; small, medium, and large, based on the size of the line break.

Following the accident initiator, normal and emergency plant systems respond to control reactivity, reactor pressure, reactor water level, and containment parameters within the design bases spectrum. Of most importance is to ensure inventory control and [illegible] sufficient heat removal from the core to prevent overheating and subsequent fuel damage. Failure to provide ~~this~~ these ~~heat removal~~ functions results in core uncovery, fuel overheating, and the potential for oxidation and melting of the reactor core.

In response to accident initiators identified through operating reactor experience and performance of probabilistic risk assessment, the NRC developed criteria for evolutionary LWRs to prevent the occurrence of such initiators from leading to a severe accident. These criteria were specified in SECY-90-016 and SECY-93-087 and include design provisions for the following: anticipated transient without scram, SBO, fires, and interfacing systems.

19.2.2.1.1 Anticipated Transient Without Scram (ATWS)

An ATWS is an anticipated operational occurrence (AOO) followed by the failure of the trip portion of the reactor protection system (RPS). AOOs are those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Dependent upon the transient and its severity, the plant may recover and continue normal operation or the plant may require an automatic shut down (scram) via the RPS. The RPS is designed to safely shutdown the reactor to prevent core damage.

These transients when coupled with a failure of the RPS may lead to conditions beyond the design basis of the plant. In these cases, the reactor must be manually scrammed in order to avoid reactor fuel damage or coolant system damage. Subsequent failure of the manual scram system and inadequate core cooling ~~may~~ would lead to core damage.

Transients with the greatest potential for significant damage to the reactor core and containment are those which lead to an increase in reactor pressure and temperature, a loss of heat sink, or a failure of the RPS to scram the reactor. During an ATWS event, reactor power, pressure, and temperature must be controlled or the potential exists for a severe accident.

The ATWS rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. For pressurized water reactors (PWRs), the ATWS rule specifies inclusion of a diverse scram system from the sensor output to interruption of power to the control rods. In Sections 7.7 and 15.3.10 of this report, the NRC concluded that System 80+ complies with the ATWS rule.

19.2.2.1.1.1 Features to Prevent and/or Mitigate

In SECY-90-016, the staff recommended that the Commission approve its position that diverse scram systems should be provided for evolutionary LWRs. In its

19.2.3 Deterministic Assessment Of Severe Accident Mitigation

19.2.3.1 Overview of the System 80+ Containment Design

The System 80+ primary containment design consists of a 61m (200 ft) diameter spherical steel shell with a nominal wall thickness of 4.45 cm (1.75 in.). This wall will be ~~thicker~~ reinforced around primary containment penetrations to structurally compensate for these openings. The primary containment encloses the nuclear steam supply system, the IRWST, SITs, the refueling canal, and associated mechanical, electrical, and HVAC support components. The spherical steel shell below the reactor cavity is protected by a minimum of .9m (3 ft) of concrete with an additional 5.5m (15 feet) of concrete below the steel shell.

The primary containment provides 94,600m^3 (3,340,00 ft^3) of net free volume and its internal structures are arranged in a manner to promote mixing throughout the containment atmosphere and accommodate the pressurization from condensible and non-condensible gas releases during severe accidents. The internal structures surrounding the steam generators are especially effective at promoting natural circulation within containment by causing a "chimney" effect. The containment contains 80 igniters to control hydrogen generated during severe accidents.

The primary containment is totally enclosed by a shield building made of reinforced concrete. The containment shield building is designed to provide biological shielding and external missile protection for the containment vessel and safety related equipment. In addition, the annulus ventilation and filtration system provides a mechanism for reducing fission product releases following severe accidents.

Steam from a reactor depressurization event is condensed in the IRWST. The IRWST is the primary heat sink and may be cooled by either the SCS or the CSS heat exchangers. Either system supplies water at about 150 psi discharge pressure. ~~This~~ delivery pressure is sufficient to function as a low pressure emergency core cooling system (ECCS) as a surrogate for the safety injection system (SIS). The normal low pressure ECCS delivers water at ~~about~~ 500 psi. between for operating PWRs

The System 80+ design does not rely on low pressure safety injection however, the SCS system.

The depressurization system is expected to reduce primary system pressure to about ~~well below~~ 150 psi. The IRWST supplies water to the CFS which provides a means of flooding the reactor cavity during a severe accident, for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases. The water flows first into the HVT through four 30 cm (12 in.) diameter MOVs and then into the reactor cavity through two 25 cm (10 in.) diameter MOVs. The IRWST also supplies water to the CSS which can be used to reduce containment temperature and pressure and remove iodine from the containment atmosphere following severe accidents.

The spherical steel containment can be vented, in the case of an internal pressurization that may challenge containment integrity, through two 8 cm (3 in.) diameter hydrogen purge vents. ABB-CE has provided this venting capability, however, they have demonstrated that venting is not needed for most of the severe accident events. For those sequences where venting would aid in limiting the containment pressure below ASME Service Level C, venting would not be needed before 24 hours into the event. The use of the hydrogen purge vent for containment pressure control is the responsibility of the technical support center. The ASME Boiler & Pressure Vessel Code was used by ABB-CE to determine the containment pressure that may be reached without exceeding ASME Service Level C. ASME Service Level C loading conditions allow material strains representative of incipient yield, assuming minimum material properties, and consequently provides a conservative estimate of the containment ultimate capacity. The pressure limits determined in accordance with ASME Service Level C criteria decrease from about 1.00 MPa (145 psia) at an average steel shell temperature of 143 °C (290 °F) to .930 MPa (135 psia) at a temperature of 232 °C (450 °F).

19.2.3.2 Severe Accident Progression

A description of the processes, both physical and chemical, which may occur during the progression of a severe accident, and how these phenomena affect containment performance, is provided in this section. This description is intended to be generic in nature; however, many aspects of severe accident phenomena depend on the specific reactor type or on the containment design features. This information has been extracted from NUREG/CR-5132, "Severe

- drainage of molten materials to the vessel lower head region (dry core scenario)

- formation of melt pool, natural circulation heat transfer, crust formation, and crust failure (wet core scenario)

- lower head breach resulting from failure of a penetration, or from local or global creep-rupture

Decay heat produced by the core must be removed to achieve adequate core cooling. Adequate core cooling can be accomplished in the System 80+ by either providing enough cooling water flow to the reactor core ~~or~~ and by removing the decay heat through the steam generators or via "feed and bleed". The mechanisms by which decay heat is removed from the reactor core include a four-train SIS with direct vessel injection, functionally interchangeable shutdown cooling and CSS, and SDS. Cooling water flow to the steam generators is provided by the main and EFWS. If the decay heat is transferred to the containment from the core it can be removed by containment heat removal systems such as the CSS and the CCWS.

In the event of failure of all safety and non-safety systems to remove the decay heat, the core will heat up to the point where damage to the fuel and fuel cladding may occur. Decay heat is transferred through the radiative, conductive, and convective heat transfer to the steam, other core materials, and non-fuel materials within the reactor. The insufficient cooling supply results in coolant boiloff and a decreasing level within the reactor vessel as the decay heat generation exceeds the heat removal rate. The coolant level within the core further decreases such that the fuel rods above the coolant level are only cooled by rising steam. The fuel rods begin to overheat and cladding oxidation in the presence of steam begins at high temperatures. As the cladding oxidizes in the presence of steam, hydrogen and additional heat are generated. The fuel cladding is made of a zirconium alloy called Zircaloy. The oxidation reaction of Zirconium cladding is as follows:

$$2H_2O + Zr = 2H_2 + ZrO_2 + 6700 \text{ J/g-Zr}$$

The initial Zircaloy oxidation involves oxygen diffusion through a ZrO_2 surface layer. As the fuel rods continue to heat up from decay heat and exothermic zirconium oxidation reaction occurs, the materials within the reactor with low melting points are expected to melt first and may form eutectics. Eutectics are mixtures of materials with a melting point lower than that of any other combination of the same components.

?

Zircaloy, with a melting point of 2030 K conversion, begins to melt, breaking down the protective ZrO_2 layer which exposes unoxidized Zircaloy. Following this, local melting of the fuel rods may cause changes in the core geometry resulting in differing steam flow paths. This can lead to an increase in the oxidation process as access to the unoxidized Zircaloy is made available; on the other hand, the melt formation or changes in the steam flow path could reduce the Zircaloy surface available for oxidation and thereby decrease the overall reaction process. In some accident scenarios where residual amounts of water remain in the bottom of the core and lower plenum, substantial steaming and oxidation can take place.

In addition to oxidation, the potential exists for the zircaloy to interact with the UO_2 fuel, forming low-melting point eutectics. Formation of eutectics may decrease the effective surface area for oxidation and overall oxidation rate. The melting point of Zircaloy is dependent upon its state and lattice structure. It has three melting points which include 2150 k (beta-Zr), 2250 K (alpha-Zr(O)), and 2950 K (ZrO_2). When partially oxidized Zircaloy is in contact with UO_2, an alpha-Zr(O)/UO_2 based eutectic will form with a liquefaction temperature of approximately 2170 K. Therefore, in the presence of good fuel/cladding contact, fuel liquefaction and melt relocation will commence around this temperature. This has the potential to affect the oxidation behavior of Zircaloy based melt.

Various severe-fuel damage (SFD) test programs sponsored by the NRC indicate that oxidation of the Zircaloy is largely controlled by the availability of a steam supply and that high rates of hydrogen generation can continue after melt formation and relocation. Some of these experiments indicate that the majority of the hydrogen generated occurred after onset of Zircaloy melting and fuel dissolution. In steam-rich experiments, oxidation took place over most of the fuel bundle length and most of the hydrogen is generated early. For steam-starved experiments, oxidation was limited to local regions of the fuel bundle and the majority of the hydrogen is generated after the onset of Zr/UO_2 liquefaction and relocation.

The System 80+ contains ~~over~~ 25,240 ~~26,500~~ kg (55,655 ~~58,500~~ lbs) of zirconium in the active fuel region which has the potential to generate over 11~~40~~00 kg (2440 ~~2520~~ lbs) of hydrogen. Hydrogen production and accumulation may represent challenges to the containment in numerous ways including deflagration, detonation, and pressurization, as hydrogen gas is non-condensible. The System 80+ containment will be equipped with 80 hydrogen igniters to consume hydrogen as it is produced during a severe accident. Because of the large containment volume, System 80+ is not threatened from pressurization of the containment from generation of hydrogen gases. The resulting pressures are well below ASME Service Level C limits.

The SFD tests indicated the potential for incoherent melt-relocation due to non-coherent temperatures within the test bundles. This is because of the different core materials present with a wide range of melting points and eutectic temperatures. Formation of eutectics would result in a nonuniform melting and relocation process. Further differences in the melt-relocation process can be attributed to asymmetric bundle heating which can increase due to Zircaloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher temperature Zircaloy will consume the available steam through oxidation at a quicker rate. The oxidation reaction increases the hotter areas to even higher temperatures, which further increases the oxidation rate and the local temperatures. This autocatalytic nature of Zircaloy oxidation appears to contribute to asymmetric bundle heatup and the potential for incoherent melt relocation behavior.

As the temperature of the core increases, the fission products are vaporized and released. These fission products are then carried by steam and/or hydrogen throughout the primary system and are subject to deposition on the surfaces of internal components. The deposition mechanisms include condensation, gravitational settling and thermphoresis. The fission products

Early containment failure

failure, the composition, amount, and character of the molten core debris expelled, the type of concrete used in containment construction, and the availability of water to the reactor cavity. The initial response of the containment from ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented by design features, early containment failure mechanisms usually dominate risk consequences. ~~These~~ mechanisms result from energetic severe accident phenomena such as high pressure melt ejection (HPME) with direct containment heating (DCH) and ex-vessel steam explosions. The long term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature due to core-concrete interaction and the availability of containment heat removal mechanisms.

and by pass

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form causing fragmentation into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the reactor cavity and into the upper containment. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. Direct attack on the steel shell is precluded in the System 80+ design because the steel shell is either protected by concrete or the crane wall. This severe accident phenomenon is known as high pressure melt ejection with DCH. To prevent this phenomenon, the System 80+ design has incorporated a reliable rapid depressurization system to provide assurance, that in the event of a core melt scenario, that failure of the reactor vessel would occur at a low RCS pressure. In the event that the reactor vessel was to fail at a high pressure, the design of the System 80+ containment provides an indirect pathway from the reactor cavity to the upper compartments of containment in an effort to decrease the amount of core debris that could contribute to DCH.

Reactor vessel failure at high or low pressure coincident with water present within the reactor cavity may lead to fuel-coolant interactions with the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from non-explosive steam generation beyond the capability of the containment to relieve the pressure such that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. ABB-CE has concluded that System 80+ is capable of withstanding the loads from the most likely fuel-coolant interaction.

The eventual contact of molten core debris with concrete in the reactor cavity will lead to core-concrete interaction (CCI). CCI involves the decomposition of concrete from core debris and can challenge the containment in various mechanisms, including: (1) pressurization due to the production of steam and non-condensible gases to the point of containment rupture; (2) the transport of high temperature gases and aerosols into the containment leading to high temperature failure of the containment seals and penetrations; (3) containment liner melt-through; (4) reactor support structures ~~(e.g., corbels)~~ melt-through leading to relocation of the reactor vessel and tearing of containment penetrations; and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors including the availability of water to the reactor cavity, the containment geometry, the composition and amount of core melt, the core melt super heat, and the type of concrete involved.

The System 80+ design has several design features to mitigate the effects of CCI. These include a CFS that can be supplied from the IRWST or externally through the CSS, and limestone based concrete for the reactor cavity floor. The CFS has been designed to provide water to assist in the cooling of core debris before it enters the reactor cavity. The CSS is capable of providing water, from the IRWST or through an external source, to control containment pressurization. Water entering the containment from the CSS will gather in the HVT where it can be directed to the reactor cavity by the CFS thereby providing an external means of flooding the reactor cavity. The limestone based concrete protects the containment liner from melt-through.

The System 80+ core contains approximately ~~26,300~~ 25,240 kg (~~58,000~~ 55,655 lbs) of Zirconium in the active fuel-clad region. Oxidation of this amount of Zirconium with steam would produce about 1,090 kg (2,400 lbs) of hydrogen. This amount of hydrogen uniformly distributed through out containment would result in a hydrogen concentration of ~~13.5~~ less than 13 percent. Therefore, to comply with 10 CFR 50.34(f)(2)(ix), the System 80+ has been equipped with a hydrogen mitigation system (HMS) composed of 80 shielded GMAC model 7G thermal igniter glow plugs. The intent of the HMS is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture. This early combustion ~~should~~ will limit the hydrogen concentration well below the 10 percent limit referenced in the rule.

The efficiency of the GMAC model 7G thermal igniter has been investigated by several experimental programs such as, The Nevada Test Site, The Hydrogen Igniter Experimental Program at Lawrence Livermore National Laboratory, and tests conducted by Fenwal, Incorporated. These programs showed that the glow plug could effectively ignite hydrogen mixtures as low as 6 percent by volume and that ignition above 8 percent by volume of hydrogen consistently resulted in complete combustion. The Fenwal tests indicated that upward burns would propagate at hydrogen concentrations as low as 4 percent by volume; at 6.5 percent by volume the burn will propagate sideways and at 8.5 percent by volume the burn propagates in all directions. The results of various experimental programs conducted at these and other facilities are summarized in NUREG/CR-5079, "Experimental Results Pertaining to the Performance of Thermal Igniters," (October 1989).

Tests of these igniters were conducted to support licensing of operating reactors with ice condenser containments and Mark III containments. ABB-CE has shown that this data base is directly applicable to the System 80+ design. Therefore, it is the staff's position that, properly placed and powered, the GMAC model 7G thermal igniter can maintain uniformly distributed hydrogen concentrations below 10 percent.

In order to ensure a highly reliable HMS, two igniters have been supplied in each subvolume in addition to adding igniters in the large upper region of the containment. A subvolume is defined as a region which has some level of air and is considered by ABB-CE to be the primary backup to the emergency diesels.

flow restriction. The redundant igniters have been divided equally into two redundant groups, A and B.

Particular attention has been paid to providing a reliable power source to the igniters for all possible conditions. All 80 igniters are capable of being powered via offsite power and the emergency diesels. This is the same as all operating plants that have igniters. However, ABB-CE has provided two additional sources to assure that power is available at all times to the igniters. The third source is a CTG. The fourth source is from batteries. In case of SBO sequences, 32 igniters can also be supplied power for a minimum of ~~up to~~ 4 hours by the Class 1E division batteries. The HMS components are non-nuclear safety related, since they are not required to prevent or mitigate the consequences of a design basis accident, but are designed to sustain seismic Category I loads.

of the detailed three dimensional visualization of the System 80+ containment layout. Additionally,

Placement of the igniters was quite involved but always came down to the critical question, "If one could develop a sequence where hydrogen could either be generated in or pass through the volume in question, then igniters should be provided." The starting point was the consideration of a series of confirmatory analyses performed by ABB-CE using the MAAP4 generalized containment model and the staff using CONTAIN. Results from these analyses were used to guide ABB-CE's placement of igniters. ~~However, they were only the starting point~~

were independently

In combination with the analyses, one also considered the possible sources of hydrogen. Two possible entry points were considered possible. Hydrogen generated in-vessel during a LOCA would be released directly into containment through the line rupture. Therefore, igniters were placed above and in the vicinity of all RCS primary piping and non-isolable connecting piping to account for these hydrogen sources. The other and more dominant pathway is associated with all transients with an intact primary system or small break LOCAs. These conditions would direct hydrogen to the IRWST via the SDS. Inside the IRWST, four igniters are located in the freeboard space above the spargers.

Any hydrogen not burned in the IRWST, because of steam inerting or a lack of oxygen to support combustion, would flow out of the IRWST through 18.6m^2 (200 ft^2) of vent area. The vents are located at the bottom of the SG that the test demonstrated a conservative depressurization rate. The observed rate is well below the rate the staff believes would be necessary to consider the possibility of creating detonable mixtures.

To further assure that rapid condensation will not occur in the System 80+ design, ABB-CE in Appendix A of the System 80+ Emergency Operating Guidelines (EOGs) states that only one containment spray train or a throttled spray train be activated when restoring sprays.

An additional consideration is the potential of generating significant concentration gradients within the containment during the course of the event. HDR experiments showed significant mixing for low elevation release points. Stratification, however, was observed for cases with an elevated release point. Therefore, particular attention was given to high release points in the System 80+ design.

The highest release point in the System 80+ would be from the pressurizer through the pressurizer house. Two igniters have been placed inside the top of the pressurizer housing and four more igniters are located outside the pressurizer. Therefore, the staff does not expect significant stratification within the System 80+ containment.

The HMS is designed to be manually actuated from the MCR. Actuation is expected upon recognition of an uncovered core. The presence of an uncovered core condition can be established by: (1) no liquid measurement in the upper plenum, as noted by the lowest RVLMS sensors, (2) core exit temperature readings above 700 °F which are indicative of superheat, and (3) SI unavailable.

The hydrogen igniter system is designed to survive a severe accident environment. This is accomplished by locating transformers and power supplies outside of containment and only having the igniter located within the containment. Power will be supplied to igniters via cables ~~mineral insulated cable~~s. Too specific ~~Cables expected to be used for the HMS will be high temperature reinforced mica, overall glass braid encased. These cables are~~ designed for operation during a 45-minute continuous burn at 650 °C (1200 °F).

The 10 CFR 50.34(f)(3)(v) requires containment integrity to be maintained below ASME Service Level C Limits for steel containments during an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction. ABB-CE performed analyses, based on the methodology described in CESSAR-DC Appendix 19.11E to determine the pressurization resulting from adiabatic isochoric complete combustion of hydrogen produced by oxidizing ~~100~~ 105 percent of the System 80+ active fuel-clad material. This was assumed to be a bounding approach. The maximum calculated containment pressure was less than 102 psia which is below the ASME Service Level C stress intensity of 135 psia at a temperature of 232 °C (450 °F).

The staff concludes that the results of these analyses show that the design satisfies this regulatory requirement.

19.2.3.3.1.2 Basis for Acceptability

The System 80+ design meets the requirements of SECY-90-016 and 10 CFR 50.34(f)(2)(ix) by providing a system for hydrogen control that provides reasonable assurance that uniformly distributed hydrogen concentrations inside containment will not exceed 10 percent. The System 80+ design is capable of withstanding the pressurization loadings associated with the complete combustion of hydrogen produced by oxidizing 100 percent of the active fuel-clad material as required by 10 CFR 50.34(f)(3)(v).

19.2.3.3.2 Core Debris Coolability

Core debris coolability and quenchability have been the subject of extensive research over the past decade; however, much uncertainty still exists relative to this phenomenon which will most likely not be resolved in the near future. Due to this uncertainty, the NRC decided that the question is not whether coolability or quenchability has been achieved or can be achieved; but rather, what is the impact on the containment design if they are not achieved.

CCI is a severe accident phenomenon that involves the melting and decomposition of concrete in contact with molten core debris. This phenomenon may occur following accident sequences which result in molten core debris process of core debris. The system would only be activated if corium melt-through of the reactor vessel appears to be probable. Once activated, the water is intended to flood the cavity floor before the core debris pours into the cavity. The water also cools and/or condenses gases evolved during CCI, thereby limiting containment temperature and pressure increases, and scrubs fission product releases. The CFS is discussed in CESSAR-DC Sections 6.8, 19.6.3.16, and 19.11.3.3.

The CFS consists of four 30cm (12 in.) diameter spillways from the IRWST to the HVT and two 25cm (10 in.) diameter spillways that connect the HVT with the reactor cavity. The CFS valves are located approximately 1.5m (5 ft) above the basemat to avoid direct core debris attack. The HVT spillways and the reactor cavity spillways are equipped with remote manually actuated motor operated valves that are qualified for submerged operation. The CFS is seismic Category I.

Each holdup volume flooding valve is powered from separate Class 1E channels and each cavity flooding valve is powered from separate Class 1E divisions. The Class 1E busses are normally supplied from offsite power sources. Upon LOOP, power to the busses can be supplied by the Class 1E emergency diesel generators or the Class 1E batteries. In addition, the CTG can power these busses upon loss of all other ac power. Once the valves have been actuated, movement of the water from the IRWST to the cavity occurs passively due to the natural hydraulic driving heads of the system. Fully flooded, the reactor cavity water level will be ~~at least~~ about 5.2m (17 ft) above the reactor cavity floor. The CFS has been designed to flood the reactor cavity to the 1.5m (5 ft) level in at least 30 minutes. The time to completely fill the reactor cavity to the equilibrium elevation was calculated to be about 72 minutes with two HVT spillway valves and one reactor cavity spillway valve open. The maximum flood level was established to avoid contact between the cavity flood water and the ICI plates below the reactor vessel lower head in case of an inadvertent actuation of the CFS.

Accident management guidance indicates that the CFS will be actuated once a potential core melt condition is imminent or has been diagnosed as being in progress. Indications capable of diagnosing core uncovery include: (1) core exit thermocouple temperatures in excess of 650 °C (1200 °F), (2) reactor vessel level monitoring system readings indicative of no liquid above the fuel alignment plate, and (3) significant changes in readings of self-powered neutron detectors.

The staff concludes that the CFS meets the criteria of SECY-93-087 relative to providing a means to flood the reactor cavity to assist in the cooling process of core debris.

19.2.3.3.2.1.3 Containment Spray System (CSS)

The CSS is a safety grade and seismic Category I system designed to reduce containment pressure and temperature and remove iodine from the containment atmosphere following a main steam line break, a LOCA, or a severe accident. The CSS provides spray of borated water to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the IRWST. The CSS is discussed in Sections 6.5 of the CESSAR-DC and Section 6.5 of this report.

The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in a reduction in containment pressure and temperature. The CSS is designed to provide adequate cooling of the containment atmosphere to limit post-design basis accident building temperatures and pressures to less than the containment design values (3.7×10^2 kPa (53 psig) and 143 °C (290 °F)). ~~The boric acid solution used to~~ ~~spray~~ Spraying the containment atmosphere minimizes the fission product iodine by the removal of iodine by the spray droplets.

Containment spray flow can also be provided ~~The CSS pumps are also capable of taking suction~~ from an external source of water via a "tee" connection. Water from the containment sprays is collected in the HVT. As described in Section 19.2.3.3.2.1.2 of this report, the CFS is designed to direct water from the HVT to the reactor cavity to assist in the process of cooling core debris.

The staff concludes that the CSS is capable of reducing containment pressure and temperature and provides another means of flooding the reactor cavity to assist in the cooling process of core debris, as specified in SECY-93-087.

depending on the aggregate selection cor

19.2.3.3.2.1.4 Sacrificial Limestone Based Concrete

are calcium carbonate based and are

Limestone ~~based~~ concretes ~~is a type of siliceous concrete~~ used in the construction of nuclear power plants and is found throughout the United States. This concrete melts over a range of 1653 to 1873 K and ~~typically~~ between about 20 liberate~~s~~ ~~30~~ to 35 weight-percent CO_2 (bound ~~as~~ as $CaCO_3$) ~~CO_2 gas~~ and 4 to 5 weight-percent H_2O. ~~vapor when heated to melting (NUREG/CR-5443)~~. Destruction of limestone based concrete releases carbon dioxide gas and water vapor.

In CESSAR-DC Section 19.11.3.6.2, ABB-CE indicated that the minimum distance between the floor elevation and the embedded portion of the containment shell is a minimum of 0.9m (3.0 ft) in the reactor cavity. Directly under the reactor vessel this distance increases to a maximum of 1.5m (5 ft). An additional 4.6m (15 ft) of concrete is available below the containment liner elevation. The basemat will be constructed of either limestone-common sand or limestone aggregate type concretes. Limestone based concrete was chosen because of its superior resistance to ablation when compared to other commonly used basemat materials such as basaltic concrete. This improved ablation resistance allows ABB-CE to maintain containment integrity without further increasing basemat thicknesses. The results of the analyses provided by ABB-CE and the staff are provided in Sections 19.2.3.3.2.2.1/2 of this report, respectively.

The staff concludes that the 0.9m (3.0 ft) layer of limestone concrete provides sufficient protection for the containment liner and that the criteria specified in SECY-93-087 relating to protecting the containment liner have been met.

19.2.3.3.2.1.5 Reactor Vessel Support Structure

The limestone based concrete discussed above protects the containment liner from core-concrete attack in the axial direction. Core-concrete attack in the radial direction could impact the reactor cavity walls. The reactor cavity

19.2.3.3.2.2.3 Conclusions

(This section will be updated to include the results of BNL analyses.)

The staff did not rely on any one specific sequence or scenario performed by ABB-CE using the MAAP 3 code nor by the staff's contractor (Brookhaven National Laboratories) in determining whether the System 80+ met the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to containment overpressurization. The CFS and CSS are fundamental to prolonging the period to containment overpressurization or melt-through of the containment liner.

The staff concludes that the System 80+ design meets the criterion when credit is given to the mitigation systems incorporated into the design, such as the CFS and CSS.

19.2.3.3.2.3 Basis for Acceptability

(This section will be updated to include the results of the BNL analyses.)

The System 80+ has met the criteria of SECY-93-087 by (1) providing an unobstructed reactor cavity floor that promotes debris spreading;
(2) providing a diverse and redundant means of flooding the cavity;
(3) providing at least a 0.9m (3.0 ft) layer of limestone based concrete to protect the containment liner; and (4) providing a robust reactor cavity~~; and (5) providing a containment vent~~. Containment conditions resulting from CCI can be maintained below Service Level C for 24 hour, through incorporation of the above listed design features.

19.2.3.3.3 High Pressure Core Melt Ejection (HPME)

HPME and subsequent direct containment heating (DCH) is a severe accident phenomenon that could lead to early containment failure with large radioactive likelihood that the valves will be available well into a severe accident, the staff has concluded that the RDS valves will be available to depressurize the RCS during a severe accident.

The design of the reactor cavity of the System 80+ is expected to decrease the amount of ejected core debris that reaches the upper containment. This decrease is anticipated through: (1) capture and trapping of some debris in the reactor cavity; and (2) impaction and removal of core debris as it is transported between the reactor cavity and upper containment.

System 80+ is equipped with an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. The reactor cavity debris chamber and exit shaft have been designed such that following a failure of the reactor vessel, high inertia corium debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path.

One possible pathway from the reactor cavity to the upper containment would be the instrument shaft. To minimize the possibility of corium carryover, the vertically oriented shaft has been provided with a limited gas venting area. Analyses performed by ABB, based on models developed by Sandia, indicate that only 10 percent of entrained corium could be expected to initially be carried upward into the shaft. Finally, gas/corium outflow into the instrument shaft is restricted by an instrument seal table which is not expected to fail upon cavity pressurization.

The reactor cavity entrance is a single stairway from the 91 feet 9 inch elevation operating deck. This stairwell connects the upper containment with the reactor cavity via a convoluted pathway through an HVAC room. The staff considers this pathway sufficiently torturous to contain ejected core debris within the reactor cavity.

19.2.3.3.3.2 Basis for Acceptability

In SECY-93-087, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization

(250 °F). This profile was used as the environmental condition associated with a global burn of hydrogen generated by the equivalent of a 100 percent metal water reaction.

The staff had several severe accident sequences analyzed with its computer model, MELCOR, to confirm the ability of the computer model used by ABB, MAAP 3, to predict the environmental conditions attendant with a severe accident. Based on this confirmation, the staff concludes that the environmental conditions predicted above by MAAP are acceptable approximations of the environmental conditions for which mitigative features and instrumentation, identified in this section, must survive.

19.2.3.3.6.2 Equipment and Instrumentation Necessary to Survive

ABB-CE identified the instrumentation and equipment required for severe accident mitigation and recovery in Tables 19.11.4.4-1 and 2 of the CESSAR-DC. The equipment listed is necessary to ensure that adequate inventory and heat removal can be provided to the RCS, reactivity control can be maintained, hydrogen can be ~~removed~~ controlled, and containment heat removal via sprays is functional. The list of equipment also includes the CFS and the containment penetrations in the case of an ex-vessel event. The instrumentation was chosen such that the operator could confirm and trend the results of actions taken and that adequate information would be available for those responsible for making accident management decisions.

The instrumentation and equipment required for severe accident mitigation and recovery will be demonstrated to operate in the applicable environment described above in Section 19.2.3.3.6.2. The demonstration process used to provide reasonable assurance that the instrumentation and equipment will operate will include one or more of the following factors: limited time period in or exposure to the environment, the use of similar equipment in commercial industry exposed to a similar environment, the use of analytical extrapolations, the use of vendor performance data, the use of procurement specifications imposed on the vendor, or the results of tests performed in the nuclear industry or at independent laboratories.

In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of integrity of the containment shell ~~structural integrity~~; and (2) releases which result in significant offsite doses. Using the dose definition of containment failure the CCFP for the System 80+ is approximately 3 percent while using the shell ~~structural~~ integrity definition of containment failure results in a CCFP of approximately 11 percent, which is slightly higher than the goal. Section 19.1.3.2.1 of this report provides the results of the CCFP analyses and concludes that because of the approximate nature of the containment performance goal, the recognition that PRA results, particularly bottom line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11 percent CCFP estimate are late, containment basemat melt-throughs rather than releases to the atmosphere. Consequently, it is concluded ~~that~~ the System 80+ design meets the Commission's containment performance goal.

which locally penetrate the shell at the bottom without compromising the typically structural integrity of the shell and release radiation into the soil

19.2.5 Accident Management

Accident management (AM) encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage; (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel; (3) maintain containment integrity as long as possible; and (4) minimize offsite releases. AM, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design-basis into severe fuel damage regimes, and by making full use of existing plant equipment and operator skills and creativity to terminate severe accidents and limit offsite releases.

Based on PRAs and severe accident analyses for the current generation of operating plants, the NRC staff concluded that the risk associated with severe accidents could be further reduced through improvements to utility accident management capabilities. Although future reactor designs such as System 80+ will have enhanced capabilities for the prevention and mitigation of severe accidents, accident management will remain an important element of defense-in-depth for these designs. However, the increased attention on accident prevention and mitigation in these designs can be expected to alter the scope, focus, and overall importance of accident management relative to that for "dry" cavity severe accident sequences. Therefore, the failure of EPAs is considered to be remote by ABB-CE.

For the mechanical penetrations, ABB-CE stated that the onset of rapid failure of seals was above 329.4 °C (625°F) provided the seal was constructed from either a ethylene-propylene (EP), neoprene, or silicone. Seal failure was defined as the inability of the seal to maintain a high (approximately 1135.56 kPa (150 psig)) containment pressure. Gradual degradation of these seals were noted at the temperature in the 148.8 to 204.4 °C (300 to 400 °F) range. The typical EP seals will require more than 20 hours to fail when subjected to a sustained period of high temperature exposure. Silicone based seals provide for even longer high temperature stability. The capability of either sealant material is sufficient to guarantee containment integrity for periods of more than 1 day for all "wet" cavity sequences. These sequences comprise more than 90 percent of the severe accident transients. Analyses performed by the staff using the MELCOR computer model showed that the temperatures do not exceed 260 °C (500 °F) for dry cavity sequences in the first 24 hours.

CESSAR-DC Section 19.11.3.1.4 states that the intent of the penetration seal design is to ensure that the selected seal and mounting will provide a minimum of 1 day containment integrity. This intent will be accomplished by a combination of selecting high quality and high capability seals, protectively mounting the seal so that it is not directly exposed to the containment environment, and providing double seals (inner and outer) whenever possible. This is judged by the staff to be achievable based on the current penetration design and is, therefore, acceptable.

basement melt-Through or

ABB-CE states that as a consequence of ABB-CE design philosophy for the System 80+, seal failures will not cause a failure of the containment prior to 24 hours and that for all "wet" cavity sequences the seal capacity is higher the containment capacity. Dry containment sequences that do not result in containment overpressure are assumed to fail due to temperature degradation of the seals. In order to estimate the consequences of a containment seal failure, the seal leakage area was estimated by ABB-CE to be 92.9 cm² (0.1 ft²).

The staff considers the treatment of penetration seals acceptable under severe accident. ABB-CE incorporates its leakage area estimate as a function of internal pressure in its MAAP analysis. The staff reviews ABB-CE's source term estimates in Section 19.2.3.3.6 of this report.

19.2.6.5 Conclusion

The staff concludes that the design of the steel containment under severe accident phenomenology will meet the deterministic containment performance goals of SECY-90-016. The conclusion is based on: (1) the evaluation of capacity using ASME Code Level C Service Limit and a 3-D finite element model analysis, (2) the realistic to pessimistic failure probability assessments for various pressure ranges, and (3) the due consideration of the effects of any potential localized leakage from thermal buckling at the transition area, at the penetrations, and at penetration seals.

The median pressure capacity should ensure that the containment would provide a reliable barrier against uncontrolled release of fission products as long as the internal pressure generated by severe accident events does not exceed 1185.89 kPa (172 psia) at 143.3 °C (290 °F) or 1103.16 kPa (160 psia) at 232.2 °C (450 °F). On this basis, the staff considers that localized leakage from thermal buckling at the transition area, from the penetrations, and from penetration seals are duly accounted for.

Comparison of the current PRA fragility values from the linear approximation to the fragility curve obtained from the combined beta method based on the lognormal distribution shows that both methods achieve reasonable results. The linearly fitted fragility curve for PRA is acceptable because the net effect of using the linearly fitted fragility curve is to produce higher DCH conditional containment failure probabilities compared to the combined beta method. This is conservative and acceptable.

close the containment hatch in the los of shutdown cooling event in Mode 5 other than reduced inventory operations.

The staff finds that ABB-CE's approach for the containment closure in Mode 5 other than reduced inventory operations acceptable and that ABB-CE has appropriately addressed concern in NUREG-1449.

19.3.7 Shutdown Risk Insights

The staff reviewed ABB-CE's shutdown risk PRA for the System 80+ design. The study addressed CDF from internally initiated events in Modes 3, 4, 5 and 6; and vulnerabilities while operating the plant in modes other than full power. The staff also considered human reliability insights, important human actions, insights from uncertainty, importance, and sensitivities analyses. Details of the PRA insights for System 80+ are discussed in CESSAR-DC Chapter 19 and Section 19.1.5 of this report.

The fundamental conclusion of the staff evaluation of the PRA-based insights for System 80+ shutdown operation is that there are no significant vulnerabilities that would require design changes. The following are considered important shutdown risk insights for the System 80+ design:

- During plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures, training, and operator knowledge of plant configuration. This issue is Section 19.3.5 of this report.

- During plant shutdown, the integrity of fire protection and flood barriers between areas in the same division, such as quadrants, where systems comprising the alternate shutdown success paths are located, should be maintained. This issue is ~~a~~ COL action item 19.18.~~, which will be documented in the flooding and fire protection, and will be provided at a later date~~.

modifications evaluated would be cost-effective given the low residual risk for the System 80+, and the $1000 per person-rem criterion.

The staff has considered the robustness of this conclusion relative to a number of critical assumptions in the analysis as described below. These involve: the effect of uncertainties in estimating core damage frequency, the use of alternative cost/benefit criterion, and the inclusion of external events within the scope of the analysis.

Based on uncertainty analyses performed by ABB-CE for the Level 1 portion of the PRA (see Section 19.1.3.1.3 of this report), the 95th percentile core damage frequency is approximately 5E-6 per reactor year. This is about a factor of 3 higher than the mean value on which the cost/benefit analysis is based, but still very low both compared to operating plants and in absolute terms. Even if the benefits of the various design improvements were requantified on the basis of this upper bound value, none of the improvements would become cost beneficial. This would remain the case even if the cost/benefit criteria was also increased by a factor of 10 to $10,000 per person-rem averted.

If external events are included, the estimate of System 80+ risk could be one ~~or possibly two orders~~ order of magnitude higher than considered in this analysis. However, even assuming the higher risk estimate and complete elimination of all risk, any design modifications or combinations which cost more than $1.7 million dollars would not be cost effective. Based on ABB-CE analysis, those modifications which were estimated to cost less than $2 million dollars have a relatively low risk reduction potential, and would generally eliminate only about 10 percent of the residual risk from internal events. The lower cost improvements are also not expected to be effective in eliminating most of the added risk from seismic events. As a result, none of these improvements are expected to be cost effective when their actual effectiveness in reducing risk is taken into account.

reduced. This is true for both internally- and externally initiated events. For example, the System 80+ seismic design basis (0.3g SSE) has been shown to result in significant ability to withstand earthquakes well beyond the design basis, as characterized by a high confidence with low probability of failure (HCLPF) value of about ~~0.6~~ 0.7g. Moreover, with the features already incorporated in the System 80+ design, the ability to estimate core damage frequency and risk approaches the limitations of probabilistic techniques. Specifically, when core damage frequencies of one in a hundred thousand or a million years are estimated in a PRA, it is the areas of the PRA where modelling is least complete, or supporting data is sparse or even non-existent that could actually be the more important contributors to risk. Areas not modelled or incompletely modelled include human reliability, sabotage, rare initiating events, construction or design errors, and systems interactions. Although improvements in the modelling of these areas may introduce additional contributors to core damage frequency and risk, the staff does not expect that they would be significant in absolute terms.

10 CFR 50.34(f)(1)(i) requires an applicant to perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. The staff concludes that the System 80+ PRA, and ABB-CE's use of the insights of this study to improve the design of the System 80+ meets this requirement. The staff concurs with ABB-CE conclusion that none of the potential design modifications evaluated are justified based on cost-benefit considerations. It is further concluded that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations, because the estimated core damage frequencies would remain very low on an absolute scale.

19.4.8 References

1. ABB-CE letter LD-92-056, dated April 24, 1992.

2. ABB-CE letter LD-93-098, dated June 18, 1993.

19.A.1 Closure of DSER Open Items

In preparing the DSER for the System 80+ PRA, the staff identified xx open items. ABB-CE provided responses to all DSER items. The staff reviewed ABB-CE's responses and in many cases resulted in request for additional information. The review by the staff of ABB-CE's responses to the DSER open items, including responses to follow-on questions, found that the applicant satisfactorily addressed these issues. Therefore, the staff considers all open items raised in the DSER to be resolved. The DSER open item closure is summarized below.

Open Item 19.1.1.1-1: At the time the DSER was prepared the applicant was updating its IRWST design. For the DSER item, the staff stated that the PRA will be revised by ABB-CE to reflect the design change and that the staff will re-evaluate the possibility of an unisolable IRWST leak due to a pipe break The applicant used the final IRWST design in the PRA-based seismic margins analysis. The staff review found ABB-CE's model of an unisolable IRWST leak to be acceptable.

Open Item 19.1.2.1.1.2-1: The staff requested ABB-CE to evaluate the potential impact of failure to open of one or more primary safety valves (PSVs) to prevent RCS overpressurization following an ATWS event. This evaluation was needed to determine the success criterion regarding the number of PSVs that must open to achieve successful RCS pressure relief during an ATWS event. In response to this open item the applicant performed a series of ATWS transient analyses to evaluate the System 80+ response to an ATWS event as a function of the moderator temperature coefficient (MTC) and the number of PSVs that must open to mitigate the pressure transient. The results of these analyses, in conjunction with other ABB-CE analyses to determine the level C stress limit pressure (i.e., the pressure level above which RCS integrity, or the operability of the systems needed for safe shutdown, can be jeopardized), were used to determine the number of PSVs that must open assuming different MTC values. To account for uncertainties in the deterministic "best estimate" analyses, and for PRA modeling purposes only, it was conservatively assumed that the level C stress limit pressure is 3200 psia. The applicant modified

Open Item 19.1.2.1.2.2-1: A number of concerns were identified in the DSER regarding the validity of several of the PDS deletion rules. In response to this open item, the applicant changed the deletion rules in question and corrected other inconsistencies discussed in the DSER. The changes were reflected in the updated PRA, and resolve the concerns identified in the staff's earlier review.

Open Item 19.1.2.1.2.3-1: Some of the parameters that are important to defining fission product release (e.g., release point) were included in the PDS definition, but were not included in the containment event tree or the release class definition. As a result, PDSs with significantly different in-vessel releases to containment could be grouped in the same release class. In response to this concern, the applicant revised the CETs and the related supporting logic models to include top events for fission product scrubbing for the various release paths (e.g., in-vessel and vaporization releases). Although there are still parameters that are important to fission product releases that are not included in the CET top events (e.g., the RCS leakage rate, which affects fission product deposition in the RCS), the staff concludes that these changes acceptably resolve the concerns raised in the earlier review.

Open Item 19.1.2.1.2.3-2: In the DSER, the staff noted that several of the CET end states in the original PRA were not necessary because they were physically impossible, and that certain additional end-states may be needed. In response to this item, the applicant corrected the CET to eliminate physically impossible end states, and added additional end states to cover outcomes not originally modelled. Based on a review of the updated CETs, the staff concludes that the problems noted with the original CETs have been eliminated.

Open Item 19.1.2.1.2.4-1: The frequency of containment isolation failure in the System 80+ PRA was taken directly from WASH-1400. In response to staff concerns regarding the applicability of this value to the System 80+ design, the applicant performed a plant-specific assessment of the probability for containment isolation failure due to piping penetration failure. The contribution to containment isolation failure from other important penetrations, such as electrical penetrations, equipment hatch, and personnel failure (see Open Item 19.1.2.1.2.4-11). The staff has reviewed these revised models and the associated containment failure probability values and finds them to be acceptable.

Open Item 19.1.2.1.2.4-11: The potential for localized containment failure due to degradation of penetration materials at elevated temperatures was not modelled in the original PRA, but has been factored into the updated PRA. The probability of a containment penetration failure (under dry cavity conditions) is assumed to be 1E-3 in the updated model. A low probability of penetration failure will be assured by a commitment under D-RAP that penetrations will be designed and sealant materials will be selected to ensure that the seal and mounting will provide a minimum of 1 day containment integrity. The staff concludes that the applicant's treatment of this issue acceptably resolves this issue~~s~~ raised in the DSER.

Open Item 19.1.2.1.2.4-12: In the original PRA, the applicant assumed that once the reactor cavity was flooded the core debris would be coolable and CCI would terminate in the cavity. Since experimental studies indicate that CCI can continue despite the existence of an overlying water pool, the staff questioned the validity of this assumption. In response to this item, the applicant revised the CCI model in the updated PRA. For the base case analysis in the updated PRA, the applicant assumed a 50 percent probability of achieving debris coolability given a wet cavity. The updated PRA also considers the potential for basemat melt-through in wet cavity cases, and assigned a 1 percent probability to this failure mode on the basis of high heat transfer rates to the water. The impact of these assumptions on PRA results was separately determined via sensitivity analyses. The staff concludes that the applicant's modelling changes and supporting analyses adequately address the concerns raised in the DSER.

Open Item 19.1.2.1.2.4-13: In the original PRA a complete basemat melt-through to the underlying stone or soil was assumed to not result in an atmospheric release and was treated as a "no containment failure" case. Since the consequences from this case would not be the same as a containment failure case, the staff requested the applicant to provide a further evaluation of the fission product releases associated with basemat melt-through and, if

Confirmatory Item 19.1.2.1.1.2-1: The staff requested documentation showing that with no secondary cooling and no SITs, a single SIS pump can prevent core damage during a medium LOCA. ABB-CE performed a transient analysis using the MAAP computer code and showed that this statement is accurate.

Confirmatory Item 19.1.2.1.1.2-2: The staff requested that the success criterion for aggressive secondary cooldown (ASC) during a small LOCA (i.e., the criterion that "all four SITs must inject borated water into the RCS during depressurization") be modeled in the fault trees. The applicant added this in the appropriate fault tree.

Confirmatory Item 19.1.2.1.1.3-1: The staff noticed that the event "failure of a primary safety valve (PSV) to reseat after opening was not modeled in the event trees developed for loss of main feedwater and "other transients (these events could be equivalent to small LOCAs). In the revised PRA, ABB-CE modified the affected event trees in accordance with the results of applicable transient analyses.

Confirmatory Item 19.1.2.1.1.3-2: The staff requested that ABB-CE reports separately the station blackout (SBO) cutsets from the rest of the LOOP cutsets (the concern was whether any important SBO sequences have been overlooked given that the staff was able to identify one missing cutset). A detailed breakdown of the LOOP cutsets into blackout and non-blackout cutsets was presented in ABB-CE Letter LD-92-113.

Confirmatory Item 19.1.2.1.1.3-3: The staff noticed that an important station blackout cutset was missing. This involved LOOP, followed by common-cause failure of the diesel generators and common-cause failure of the turbine-driven EFW pumps. This was included in the revised PRA.

Confirmatory Item 19.1.2.1.1.4-1: The staff was unable to solve 2 of 67 top level-functional fault trees using the IRRAS computer code. The applicant investigated these two fault trees and supplied new versions to the staff along with a discussion of what was changed and what effect this had on the risk profile.