ABB

February 24, 1994
LD-94-015

Docket No. 52-002

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

Subject: System 80+™ Information for Issue Closure

Dear Sirs:

The attachments to this letter provide revisions to CESSAR-DC and material to close follow-on questions to DSER responses. Attachment 1 provides revisions to the Structural Design Criteria of Appendix 3.8A, which should be given to Mr. G. Bagchi.

Attachment 2 transmits a revision to Section 3.11 to reflect resolution of NRC questions on implementation of the new radiological source term in the equipment qualification program. These revisions should be given to Mr. H. Walker and Table 3.11-2 showing the Safety Injection System in the higher Level 2 qualification group should be given to Mr. T. Collins.

Attachment 3 is a copy of a February 22 fax to Mr. S. Sun with minor changes to Table 4.4-1.

Attachment 4 provides revisions to Section 9.1 recently requested by NRC and a notation (not a revision) to Figure 9.1-3 to show connections for the borated and non-borated makeup water connections.

Attachment 5 transmits revisions to Section 9.4, on ventilation systems, which should be given to Mr. J. Raval.

Attachment 6 presents revisions to Section 14.3 based on recent discussions with Mr. T. Boyce on the issue of design detail provided and the use of ITAAC This issue has at times been generally identified as Design Acceptance Criteria, although the enclosed revisions do not use this terminology.

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Attachment 7 provides minor revisions to Chapter 7 to clarify the use of modified statistical combination of uncertainties methodology. This clarification was made as a result of the February 9 ACRS Subcommittee meeting.

Attachment 8 presents revisions to CESSAR-DC and the Emergency Operations Guidelines resulting from the resolution of the Shutdown Risk (Mode 5 Drain-Down) Operator Action Time issue. These revisions should be given to Mr. T. Collins.

Attachment 9 provides revisions to the list of COL License Information (action items) in Section 1.10. This section will be revised again after the System 80+ Advanced FSER is issued.

Attachment 10 transmits revisions to Chapter 19 which are nearly identical to those faxed to Mr. N. Saltos on February 14.

Attachment 11 transmits an overview and the quality plan which implements the QA Topical Report referenced in Chapter 17 of CESSAR-DC. ABB-CE agreed to make this submittal at the February 18 closeout interview after the QA audit by NRC staff.

CESSAR-DC changes provided above will be printed in Amendment V.

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

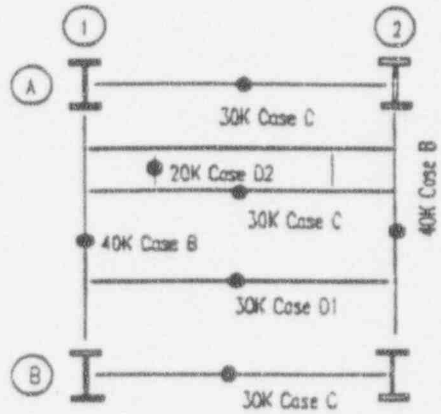
C. B. Brinkman
Acting Director
Nuclear Systems Licensing

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

ATTACHMENT 1

5.1.1.2.7 Miscellaneous Equipment and Large Bore Piping

The following load allowances shall be considered where multiple large bore piping runs are located or where large temporary loads are identified.



- In addition to major equipment located on general arrangement drawings, a point load of 20 kips should be applied at the midpoint of each concrete floor slab and concrete beams (Case A).
- A point load of 40 kips shall be applied at the midpoint of steel collector beams providing primary framing (Case B).
- A point load of 30 kips shall be applied to the midpoint of other steel collector beams or beams provided for support framing (Case C).
- A point load of 30 kips at midspan on primary steel filler beams framing into steel collector beams (Case D1) and 20 kips on other steel filler beams or stringers (Case D2). (Note: These loads are for added design margin on the beams and slabs and are not to be carried beyond the beam support connection to the supporting beam or column.)
- A contingency load of 80 kips should be applied to the top of each steel column.

5.1.1.2.8 Miscellaneous Equipment, Small Bore Piping, Cable Tray, and HVAC Ductwork

The following load allowances should be included for areas with multiple runs of small bore piping, cable tray, or HVAC ducts.

- A load of 15 kips on steel collector beams
- A load of 5 kips on other steel beams
- A load of 50 kips on steel columns

Add attached 5.1.1.2.9 insert

5.1.1.3 H - Soil Load

Lateral soil pressure shall be based upon the soil density and shall include the effects of ground water in accordance with section 5.1.1.4 of this appendix. Normal soil loads shall consider a ground water level up to El. 88'-9", 2'-0" below plant finished yard grade elevation (El. 90'-9"). The lateral soil pressure shall be based upon the following soil properties;

- Soil Density
  125 pounds per cubic foot (pcf), normal moist soil
  80 pcf, dry
  145 pcf saturated

Insert 5.1.1.2.9

5.1.1.2.9 Alternate Load Allowances for Piping, Cable Trays, Conduit, HVAC Ductwork and Miscellaneous Equipment

The following alternate load criteria may be used in lieu of Sections 5.1.1.2.6, 5.1.1.2.7 and 5.1.1.2.8.

- For piping, cable trays, conduits and HVAC ducts, conservative estimated loads shall be used with a minimum value of 50 psf.
- For major equipment, actual loads shall be used
- In addition to the above loads, 5 kips concentrated load on beams, girders and slabs shall be used to maximize moment and shear. This load is not carried beyond the beam support connection to the supporting beam, girder or column.

Actual loads shall be tracked during the design process, reconciled with the load allowances established and documented in the structural analysis report described in the structural acceptance criteria, Section 3.8.4.5.3

Replace with attached 6.2.1.1.1 insert

- Transverse reinforcing at the edges of wall panels shall be anchored in accordance with Paragraphs 21.5.3.5 and 21.5.3.6

- Longitudinal reinforcing for beams shall be anchored according to Paragraph 21.6.1.3 with hoop reinforcement per Paragraph 21.6.2.1

- Development lengths for reinforcing will be according to Paragraph 21.6.4.

~~Epoxy coated reinforcing shall be used for exterior walls and slabs when the existing groundwater is determined to be sufficiently corrosive so as to adversely affect the long term durability of the concrete structure. The required splice length given in ACI 349 Section 12.2.2 shall be increased using factors provided in ACI 318 Section 12.2.4.3.~~

Y

When feasible, uniform reinforcement patterns should be used for sections with similar requirements, thickness and loading.

6.2.1.1.2 Concrete Expansion Anchors

Expansion anchors shall be of the wedge, sleeve, or undercut design as specified in Section 3.8.4.5. Minimum design safety factors shall be:

- 4.0 for wedge and sleeve type anchors
- 3.0 for undercut type anchors

Expansion anchor embedments shall have a minimum factor of safety of 1.5 for concrete failure with respect to anchor minimum tensile strength.

Selection of expansion anchors shall consider energy absorption capability (i.e. ductility) of the anchors.

A specification for the design, installation, and use of expansion anchors should be developed by the COL Applicant and include;

- expansion anchor allowable loads,
- expansion anchor minimum spacing,
- spacing requirements for expansion anchors,
- procedures for addressing baseplate flexibility's in calculating design loads on expansion anchors,
- procedures for addressing shear tension interaction, and
- required load reductions for cyclic loadings.

When high capacity concrete anchors are specified, they should be of the direct bearing or "undercut" type. Load transfer for these anchors is achieved by bearing of the expanded embedded tip against the undercut concrete hole produced by a special flaring tool. Undercutting of the concrete is required for the anchor to provide the concrete shear capacity to match the high strength bolts.

For smaller safety related or non-safety related applications expansion anchors referred to as "Sleeves" or "Wedges" may be used, subject to the safety factors given above.

Insert 6.2.1.1.1

Unless the ground water level is below the goundation level, due to either natural site conditions or provision of a permanent dewatering system by the COL Applicant, epoxy coated reinforcing shall be used for exterior walls and slabs when the existing groundwater is determined to be sufficiently corrosive so as to adversely affect the long term durability of the concrete structure. When epoxy coated reinforcing is used, the required splice length given in ACI 349 Section 12.2.2 shall be increased using factors provided in ACI 318 Section 12.2.4.3.

7.1.2 CONCRETE PLACEMENT

Requirements and/or limitations on concrete placement will be determined in conjunction with the construction schedule. A site specific construction specification should be prepared by the COL Applicant to address requirements and procedures for concrete placement.

The concrete specification should address;

- desired volume of concrete pours and rate of deposition,
- special forming requirements,
- maximum height of pours,
- temperature limitations; weather conditions and concrete mix, including approved methods for temperature control, and
- curing requirements and procedures.

7.1.3 REINFORCING

Fabrication and placing of reinforcing bars for concrete in Seismic Category I structures shall conform to the requirements and tolerances specified in ACI 349 Section 7.5 and in ACI 301 Sections 5.5, 5.6, and 5.7.

Consideration shall be given for modular assemblies of reinforcing. Such assemblies shall be designed to be moved without changing their alignment.

Lap splices shall be prohibited for locations with tension stresses normal to the plane for the splice and for bar sizes greater than #11, except as provide by ACI 349 Section 12.14.2.1.

Welding of reinforcing shall be prohibited except as provided for in approved splice details.

7.1.4 CONSTRUCTION SEQUENCING

Construction sequence will be determined by the COL Applicant. Additional design requirements due to the construction sequence will be determined by the COL Applicant during the final design.

Add attached 7.1.4 insert

7.2 STRUCTURAL STEEL

7.2.1 STRUCTURAL STEEL; FABRICATION AND ERECTION

Fabrication and erection of safety related steel members shall be in accordance with AISC N690, Sections Q1.23 and Q1.25. Additional requirements are applicable as provided for in this appendix.

7.2.2 HIGH STRENGTH BOLTED CONNECTIONS

Bolts shall be installed and tightened in accordance with Section 8(d) of "Specification for Structural Joints Using ASTM A325 or A490 Bolts." The use of "load indicator" bolts or washers should be used where possible. "Snug tight" installation of bolts in "slip critical" connections shall not be permitted.

Insert 7.1.4

Advanced construction methods, such as modular construction or forming concrete slabs using metal deck, steel beams and columns which may be used to facilitate the construction sequence, which will affect design details must be justified by as-built analyses and results documented in the structural analysis report described in the structural acceptance criteria, Section 3.8.4.5.3.

7.2.3 WELDED CONNECTIONS

Welding activities associated with Seismic Category I structural steel and their connections shall be accomplished in accordance with written procedures and shall meet the requirements of ANSI/AISC N690 Section Q1.17.

8.0 STRUCTURAL ACCEPTANCE CRITERIA

Structural Acceptance Criteria are specified in Section 3.8.4.5.

Separation Criteria for Seismic Category I and non-Seismic Category structures and components shall be verified.

9.0 MATERIALS

9.1 GENERAL

Material shall conform to requirements for Section 3.8.4.6.1 and this appendix.

Materials used should be selected based upon a proven record of service in other nuclear facilities. Materials shall be specified based upon approved codes and standards. Additional material restrictions or requirements may be added by the design engineer to meet anticipated design or field conditions.

With suitable qualification and no applicable material restrictions, substitute materials may be used.

Materials used shall be qualified to withstand environmental conditions for normal and accident conditions. Site specific design specifications prepared by the COL Applicant should identify required qualifying environmental conditions.

9.2 SPECIFICATIONS

The materials identified below and in Section 3.8.4.6.1 shall be considered acceptable for the analysis and design of System 80+ Standard Plant structures.

Additional materials may be added to this criteria when qualified by appropriate codes and standards.

9.2.1 CONCRETE

~~Concrete - compressive strength = 4000 psi~~
~~(5000 psi for the Nuclear Island superstructure)~~
~~Normal weight concrete with a density of 135 to 160 pcf.~~

Replace with attached 9.2.1 insert

Cement - material shall conform to ASTM C 150 per ACI 349 par. 3.2. Cement shall conform to Type I or Type II designations except where additional qualifications are conducted for special applications.

Insert 9.2.1

Normal weight with a density of 135 to 160 pcf.

Compressive strength = 4000 psi

- Nuclear Island basemat
- Non-Nuclear Island structures

Compressive strength = 5000 psi

- Nuclear Island superstructure
  (A concrete strength of 4000 psi may be used when justified by as-built analyses and design details with results documented in the structural analysis report described in the structural acceptance criteria, Section 3.8.4.5.3)

9.2.2.3 Welding

Welding materials shall conform to the requirements of the Structural Welding Code (AWS-D1.1). AWS D1.1 Table 4.1.1 shows the compatibility of filler metal with base metal. ANSI/AISC N690 provides supplemental information on weld materials for stainless steel.

9.3 RESTRICTED MATERIAL

The use of the restricted materials should be based upon a proven need and avoided where possible. Materials that are restricted include;

- Use of teflon based low friction sliding bearing plates such as "Flurogold" or neoprene based gaskets, seals, or bearings shall be kept to a minimum due to presence of fluoride or chloride ions and the increased potential for stress corrosion cracking.

- Low melting point metals (lead, zinc, etc.) have been identified for their deleterious effect on corrosion resistance and ductility of metallic components. Restrictions on zinc will also mean a restriction on galvanized materials. This restriction is particularly applicable inside Containment where the zinc in the galvanized coating can result in chemical reactions producing additional hydrogen.

10.0 SUPPLEMENTAL DESIGN CRITERIA FOR NUCLEAR ISLAND, CATEGORY I AND II STRUCTURES

All structures located on the Nuclear Island are Seismic Category I, Safety Class 3, and Quality Class 1. Refer to Figure 3.8A-1 of this appendix for location of structures addressed in this section.

10.1 STRUCTURAL FOUNDATION/BASEMAT

10.1.1 DESCRIPTION

The Basemat is a 10 foot thick reinforced concrete slab that supports the Nuclear Island structures. The Basemat measures 334 feet by 442 feet, which includes an extension of four feet beyond the Nuclear Island perimeter along all four sides.

Add attached 10.1.1 insert

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10.1.2 DESIGN REQUIREMENTS

The basemat is designed for the envelope of reactions considering all soil cases. The basemat analysis provides support reactions assuming a homogeneous foundation subgrade. These reactions are used to determine an effective soil bearing pressure under the basemat. Reactions are represented by vertical soil springs. Spring constants are calculated based upon contributory areas and the underlying soil stiffness.

The basemat shall use a symmetrical reinforcing configuration based on the maximum required reinforcing, either top or bottom of the basemat to account for differential settlement.

Insert 10.1.1

---- to allow for one method for detailing of reinforcing at the edge of the basemat.

The four foot extension of the basemat is not credited in any analyses. Alternate design details that meet the ACI Code requirements may be used provided that the as-built design details are documented in the structural analysis report described in the structural acceptance criteria, Section 3.8.4.5.3.

Typical reinforcing details for alternate designs are shown in Appendix 3.8B, Figures 3.8B-3 and 3.8B-4.

ATTACHMENT 2

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The design criteria with respect to environmental effects on the electrical and mechanical equipment of the Reactor Protective System and the Engineered Safety Feature systems to ensure acceptable performance in all environments (normal and accident) depend upon equipment location and function. Such equipment is qualified to meet its performance requirements under the environmental and operating conditions in which it will be required to function and for the length of time for which its function is required. As far as practical, equipment for these systems is located outside the containment building in a mild environment. If this is not practical, the equipment is qualified for the environment in which it is required to operate.

A. For operation under normal conditions the systems are designed and qualified to remain functional after exposures to the following ranges of environmental conditions:

1. Temperature ranges given in Appendix 3.11A.

2. Relative humidity ranges given in Appendix 3.11A.

3. Pressure ranges given in Appendix 3.11A.

4. Expected integrated radiation exposures for 60 years given in Appendix 3.11A.

B. In addition to the normal environment, the mechanical and electrical components required to mitigate the consequences of a design basis accident (DBA) or to attain a safe shutdown of the reactor are designed to remain functional after exposure to the environment anticipated following the specific DBA which they are intended to mitigate. Anticipated environmental conditions and requirements are listed below.

1. The temperature, pressure, and humidity ranges following the design basis accidents such as the loss of coolant accident (LOCA), the main steam line break (MSLB) or "worst case" combined (LOCA & MSLB) are indicated in Appendix 3.11A.

2. The time integrated "worst case" post-accident radiation doses are indicated in Appendix 3.11A. Equipment will be designed for the types and levels of external radiation associated with normal operation plus the external radiation associated with the limiting design basis accident (DBA) for which it provides a safety function and for the length of time Add both during and after the accident for which it is required to be

functional. If more than one type of radiation is significant, each type may be considered separately.

The COL applicant will make the specific details of the plant specific environment qualification program available for NRC evaluation.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Appendix 3.11B lists the equipment required to mitigate a DBA or to attain a safe shutdown. Specific equipment for each system is discussed in the appropriate section of the Safety Analysis Report as referenced in Appendix 3.11B. The major component categories, such as motor-operated valves, pump motors, instrumentation and pressure boundary equipment and their location by area are also provided.

3.11.2 QUALIFICATION TESTS AND ANALYSES

Qualification of electrical equipment for System 80+ will comply with 10 CFR 50.49 as described below:

A. Environmental qualification of electrical equipment located in harsh environments within Combustion Engineering's scope of supply will be in accordance with the methodologies outlined in CENPD-255-A, Rev. 3 (Reference 1). Reference 1 has been reviewed and approved by the NRC staff as an acceptable methodology for environmental qualification of nuclear steam supply system safety-related electrical equipment. The NRC's approval of Reference 1 includes references to Amendment 9 of CESSAR-F. The development of CESSAR-DC necessitated a review of Reference 1 which resulted in a review of Section 3.11 of CESSAR-F, Section 3.11 and appendix J of supplement No. 3 to NUREG-0852 (Safety Evaluation Report for CESSAR System 80), and the NRC's approval letter (the NRC approval letter is included as an integral part of Reference 1). As a result of these reviews, ABB-CE has determined that there is no basis for including references to Amendment 9 of CESSAR-F in the review of CESSAR-DC. Therefore, the intent of CESSAR-DC is to incorporate CENPD-255-A, Rev. 3 by reference, independent of references to Amendment 9 of CESSAR-F. In instances where CESSAR-DC and CENPD-255-A, Rev. 3 differ, CESSAR-DC takes precedence.

B. Environmental qualification of electrical equipment located in mild environments within Combustion Engineering's scope of supply will be in accordance with the methodologies outlined in NPX80-IC-QG790-00, Qualification Guidelines for Instrumentation and Controls Equipment for Nuplex 80+ (Reference 2).

C. Environmental qualification of electrical equipment outside of Combustion Engineering's scope of supply will be in accordance with IEEE 323-1974 and Regulatory Guide 1.89, Rev. 1.

Environmental qualification of mechanical equipment will comply with GDC 1 and 4 and Appendix B to 10 CFR 50 (Criteria III, "Design Control," and XVII, "Quality Assurance Records") and will include the following:

A. Identification of safety-related mechanical equipment located in harsh environments, including required operating times;

B. Identification of non-metallic subcomponents of this equipment;

C. Identification of the environmental conditions for which this equipment must be qualified;

D. Identification of non-metallic material capabilities; and

E. Evaluation of environmental effects.

3.11.2.1 Mechanical and Electrical Component Environmental Design and Qualification for Normal Operation

Equipment which, due to its location, is not significantly affected environmentally by the DBA is said to exist in a mild (normal plus abnormal service conditions) environment. The qualification of equipment in a mild environment is taken from Qualification Guidelines for Instrumentation and Controls Equipment for Nuplex 80+ (Reference 2), rather than IEEE Std. 323-1974, which does not distinguish between mild and harsh environments. For this equipment, if no significant aging mechanism at mild conditions can be identified a qualified life is not required. This applies to both electrical and mechanical equipment. If the predicted life based on experience, aging analysis, or tests is less than the design life of the plant, that equipment is subjected to a surveillance program and a preventative maintenance program that restores it to qualified operability. The detailed maintenance/surveillance program for specific plants will be developed based on the specific equipment for that plant and the results of qualification testing and analysis for that equipment. This program is the responsibility of the owner-operator.

Appendix 3.11A provides the ranges of the design temperatures, pressures, and humidities, and radiation for typical mild environment areas in which safety-related equipment listed in Appendix 3.11B is located.

3.11.2.2 Mechanical and Electrical Component Environmental Design and Qualification for Operation During and After a Design Basis Accident

Add both during and

Equipment listed in Appendix 3.11B is designed to remain functional in the environment that exists at the equipment location during and after the design basis accident in question (e.g., LOCA and MSLB) for the time frame after the accident for which it is required to be functional, and the integrated radiation dose during normal operation. The temperature, pressure, and humidity environment inside the containment after a LOCA and MSLB is discussed in detail in Sections 6.2.1.3 and 6.2.1.4. The containment spray characteristics are given in Section 6.2.2.1. The "worst case" integrated post-accident radiation dose for those areas at which equipment is located is given in Appendix 3.11A.

The General Design Criteria, Appendix A to 10 CFR 50, are met as follows:

- Criterion 1 - Quality Standards and Records: refer to Section 3.1.1.
- Criterion 4 - Environmental and Missile Design Basis: refer to Section 3.1.4.
- Criterion 23 - Protection System Failure Modes: refer to Section 3.1.19.
- Criterion 50 - Containment Design Basis: refer to Sections 3.1.43 and 6.2.1.

The requirements of Quality Assurance Criterion III, Appendix B to 10 CFR 50, are met as discussed in Chapter 17.

The recommendations contained in the documents discussed below in A through H are utilized.

A. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment."

B. Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants." A description of the tests and analysis by which active valves are qualified is provided in Section 3.9.2.2.

C. The qualification methods and documentation requirements of IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and "Category 1" of NUREG-0588, are discussed in Reference 1. Exceptions are noted in Sections 3.11.2 and 3.11.2.1.

D. Passive pressure boundary components inside the containment are designed for the appropriate temperature and pressure environment in accordance with the applicable code to which the component is constructed. Environmental Qualification testing is not considered necessary for such components.

E. Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants."

F. Regulatory Guide 1.63, "Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants."

G. Type tests to ensure acceptability for use in the containment post-accident environment are performed for each type of cable in accordance with IEEE Standard 383-1974, "Standard for Type Tests for Class 1E Cables, Field Splices and Connections for Nuclear Power Generating Stations."

H. The materials used in the fabrication of mechanical and structural components inside the containment are selected so as to minimize corrosion and hydrogen generation resulting from contact with spray solutions. The use of aluminum and zinc is minimized in these components.

AGING FOR HARSH ENVIRONMENT EQUIPMENT

Equipment which is located in zones susceptible to a harsh environment are also exposed to a mild environment preceding the DBA. Such equipment will undergo an aging analysis that focuses on the identification of aging mechanisms that significantly increase the equipment's susceptibility to the design basis accidents. If no known significant aging mechanisms are found, a surveillance/preventive maintenance (S/PM) program will be developed to monitor for degradation. If an aging mechanism is found that is known to significantly degrade the equipment, that mechanism will be analyzed to determine whether an accelerated aging program or a periodic part replacement program is appropriate.

RADIATION FOR HARSH AND NON-HARSH ENVIRONMENT EQUIPMENT

Equipment is designed for the types and levels of radiation associated with its location and includes the normal operation contribution plus the radiation associated with the limiting Design Basis Accident (DBA) for which it is required to be functional and for the duration of time after the accident for which it is required to be operational. The levels defined in Appendix 3.11A are "worst case" values and are intended to represent an upper bound dose value for that region.

Add both during and

Equipment which is exposed to radiation equal to or above 10^4 Rads (equal to or above 10^3 Rads for electronic equipment) will be irradiated to its anticipated Total Integrated Dose (TID) prior to type testing unless determined by analysis that radiation does not affect its ability to perform its required function. Where the application of the accident dose is planned during DBA testing, it need not be included during the aging process.

A total integrated dose of less than 10^4 Rads (less than 10^3 Rads for electronic equipment) will not affect the strength or properties of material used; hence, further qualification analysis and tests for components which will be exposed to lesser radiation are not necessary.

Mechanical and electrical equipment will be qualified to appropriate radiation environments as discussed previously. If more than one type of radiation is significant, each type may be applied separately.

Gamma

Electrical equipment will be tested to gamma radiation levels developed as discussed previously. Upper bound dose values for various plant regions are presented in Appendix 3.11A.

Beta

Equipment exposed to beta radiation will be identified and an analysis will be performed to determine if the operability of the equipment is affected by beta radiation ionization and heating effects. Qualification is performed by test unless analysis demonstrates that the safety function will not be degraded by beta exposure. Equipment will be tested and/or analyzed to the beta radiation levels defined in Appendix 3.11A. Credit will be taken for available shielding. ~~(e.g., cable jackets for cable qualification)~~. Where testing is recommended, a gamma equivalent radiation source will be used. delete

delete

Neutron

Equipment exposed to neutron radiation will be identified and neutron radiation levels defined. When actual neutron dose qualification testing is not performed, an equivalent gamma radiation dose will be used for qualification testing to simulate neutron exposure. The basis for establishing an equivalent gamma radiation dose will be provided.

Chemical Spray

After a postulated accident, such as the LOCA or MSLB, components located in the containment building may be exposed to a chemical spray. Equipment is environmentally tested to these conditions and performance requirements demonstrated during and after the test. The most severe spray composition is determined by single failure analysis of the spray system. Corrosion effects due to long term exposure will be addressed, as appropriate.

Where qualification for chemical spray environment is required, the simulated spray will be initiated at the time shown in Appendix 3.11A.

Typical values of chemical spray composition, concentration and pH are defined in Appendix 3.11A, Table 3.11A-1.

3.11.3 QUALIFICATION TEST RESULTS

3.11.3.1 Instrumentation and Electrical Equipment

Qualification testing and analyses of instrumentation and electrical equipment are discussed in Reference 1.

3.11.3.2 Mechanical Equipment

Mechanical equipment is relatively insensitive to environmental conditions considering that service conditions usually far exceed environmental conditions. For mechanical equipment the service requirements and the environmental requirements are fully defined in the design specifications. The equipment designer selects materials based on extensive testing and long-time service which is compatible with the requirements. Quality assurance of design and quality control of processes assure that the component meets the specification requirements. Further, the design/manufacturing organization certifies compliance. In-service surveillance and maintenance programs, followed by refurbishment or replacement of parts if necessary, is further assurance that the safety equipment is operable.

The evaluation of environmental adequacy of equipment is initiated by the full definition of environmental requirements in equipment specifications, as stated above. Test reports and analyses which substantiate operability after exposure to the environment, and the quality assurance documentation, will be filed by the owner-operator and available for staff audit, as discussed in Section 6.0, Documentation, of CENPD-255-A, Rev. 3.

Lists identifying the components of mechanical equipment and a bill of materials will be available in accordance with CENPD-255-A, Rev. 3.

3.11.4 CLASS 1E INSTRUMENTATION LOSS OF VENTILATION EFFECTS

The need for the HVAC systems and the design bases which prevent the loss of essential ventilation are described in Sections 6.4 and 9.4. In general, the two division concept of this plant provides 100% redundancy of all essential equipment and the HVAC system. In event of a failure of one system to deliver the desired conditioned air, the second system will be energized automatically in its place. This changeover can also be achieved manually.

All of the areas under consideration are cooled with chilled water which will ensure a humidity level below that for which the equipment is qualified. Therefore, only temperature switches are provided in each room.

Table 3.11-1 lists all of the areas under consideration and the alarm provided. Also, each area is noted as safety-related or as non-safety ventilation.

Note that in the Control Building, the Vital Instrument and Equipment, Essential Switchgear Room, Essential Battery Room and Remote Shutdown Panel are served from a single unit with 100% redundancy of the HVAC unit in one division.

All of the subsphere rooms are served by a common ventilation system with a filtered exhaust. There is 100% redundancy of all components in the other Division. Each of the individual rooms in the subsphere has an individual cooling unit which picks up the additional heat created during operation. This gives 100% redundancy in each division.

The containment has standby cooling and ventilation equipment for each of the component parts of the ventilation system to maintain normal equipment qualification conditions. The containment ventilation systems are not credited in maintaining post-accident environmental conditions.

The chilled water system is divided into two separate circuits, normal and essential, in each division. A chiller and pump serve the essential units in each division. Also, a heat exchanger with pump is arranged to permit the essential circuit to be cooled by the water from the normal chiller and its pump. This permits the essential chiller to remain on standby during normal operation. The heat exchanger pump also can be used as a backup for the essential chiller pump. The chilled water temperature in the essential circuit ranges from 45° to 55°F which provides humidity and temperature control at each unit.

The main steam valve houses are open to natural circulation of outside air; therefore, no ventilation system is required to maintain normal or post-accident environmental conditions.

Class 1E equipment which is located in the control room or similar areas includes the following:

A. Plant Protection System (PPS)

B. Main Control Panels

C. Auxiliary Process Cabinet (APC)

Other instrumentation, such as process transmitters and signal converters and the Reactor Trip Switchgear System circuit breakers, are located in the Nuclear Annex or containment building. Equipment in these areas is qualified for the maximum expected temperature, radiation, humidity, and pressure under which and the duration after the accident for which the equipment is expected to be functional. both during and Add

3.11.5 CHEMICAL SPRAY, HUMIDITY, SUBMERGENCE, AND POWER SUPPLY VOLTAGE AND FREQUENCY VARIATION

3.11.5.1 Chemical Environment

Engineered Safety Feature systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in Sections 3.11.1, 6.2 and 6.3. In addition, components of ESF systems inside the containment are designed to perform their safety-related functions in the presence of the existing chemical environment, resulting from the boric acid recirculated through the Safety Injection System (SIS) and Containment Spray System (CSS). The SIS is designed for both the maximum and long-term boric concentration and pH. These chemical environment conditions are given in Appendix 3.11A.

3.11.5.2 Humidity

Equipment that may be adversely affected by a high humidity environment and required to operate in a high humidity environment but not subjected to a steam environment during DBE testing will be environmentally qualified by type test. The equipment is tested prior to the application of the high humidity environment to establish a baseline; then tested while exposed to a humid environment that envelopes the required humidity condition; and again tested after removal of the high humidity environment for comparison to the original baseline measurement. The comparison of the baseline tests determine if any degradation is present and ensures operability criteria are met.

Equipment that is subjected to steam environments will be subjected to the appropriate test profile in Appendix 3.11A.

3.11.5.3 Submergence

Equipment locations and operability requirements are reviewed to establish whether or not specific equipment could be subject to submergence during its required operating time. Flood levels both inside and outside containment are reviewed and potential impacts on equipment qualification appropriately addressed. Where operability during submergence is required, qualification will be demonstrated by type test.

3.11.5.4 Power Supply Voltage and Frequency Variation

Power supply voltage and frequency variation is addressed in the equipment design and verification process. During the design process, the range of power supply variation is determined. Equipment specifications incorporate the ranges to ensure acceptable operation. Type testing of the equipment at the extremes of power supply variations is performed if required.

3.11.6 RADIATION ENVIRONMENTAL QUALIFICATION

Safety related components are designed to ensure acceptable performance, taking into consideration normal operational radiation exposure in addition to the single most adverse post accident environment for which they are required to be functional.

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

Add from equipment installation

delete

- The radiation environment expected at component location ~~prior to and~~ up to the time the equipment is required to remain functional post accident, and
- The limiting design basis accident for which the component provides a safety function.

Radiological Source Terms

Normal operation radiation environments are developed based on the design source terms provided in Sections 11.1 and 12.2.

Post-accident radiation source terms are developed on the basis of the event to be mitigated. Such events can be divided into two general classes: LOCA and non-LOCA events, with the CEA Ejection event being viewed as a special case of LOCA with simultaneous reactivity insertion. For non-LOCA events such as main steam and feedwater line breaks the source terms are developed based on conservative estimates of fuel assembly gap fission product gas releases (see Chapter 15) and the maximum reactor coolant specific activities as discussed in Section 11.1.

For the LOCA events there are two levels of fuel damage considered. One level corresponds to a worst-case, non-fuel melting event involving decay power heatup of the fuel assemblies to a point where one hundred percent of the gap activity is released. The maximum reactor coolant specific activities discussed in Section 11.1 are also included. The timing of the coolant and gap activity releases is taken from Draft NUREG-1465 (Reference 3) with one exception; that is, 20 percent of the gap activity is assumed to be released instantaneously rather than according to the uniform release rate assumption in Draft NUREG-1465. The purpose of this assumption is to cover the activity release assumption of the CEA Ejection event discussed in Section 15.4.8. Indeed, this level of fuel damage is intended to significantly bound all accidents covered in Chapter 15 with the exception of the accident postulated to satisfy 10 CFR 100.

The second level of fuel damage is that corresponding to 10 CFR 100 which calls for a postulated design basis accident (DBA) involving substantial melting of the fuel. Consideration of this event provides defense-in-depth because it simply ignores the substantial capability and reliable design of the Safety Injection and Emergency Feedwater Systems to cover and cool the core, even under LOCA conditions. For this type of event the maintaining of fuel integrity (prevention of core damage) is no longer an issue. The issue for this level of core damage is the maintaining of containment integrity. The required manual operation of the safety-related and redundant Safety Depressurization System (SDS, described in Chapter 6) prior to core uncovery means that the primary system will be depressurized before core damage occurs. ~~Therefore,~~ The long-term cooling of core debris in-vessel ~~will~~ does not require use of the ~~Safety Injection System or the~~ Emergency Feedwater System. Insert A

In-vessel debris coolability for the 10 CFR 100 DBA will be maintained by the ~~Containment Spray~~ Safety Injection and Shutdown Cooling Systems ~~using the remote manually operable cross-connects that permit draining of coolant from the IRWST, passage through the system heat exchangers for containment heat removal, and injection into the vessel~~. To ensure proper transition from the arresting of core damage ~~and spray removal of radioactive materials from the containment atmosphere~~ to the long-term cooling mode described above, ~~the Safety Injection System and~~ the Emergency Feedwater System will be qualified for 72 hours of operation even with the 10 CFR 100 source term having been released to containment. As discussed in Chapter 15, this 10 CFR 100 DBA source term is based on the coolant and gap activity releases described above plus the early in-vessel fuel melt release from Draft NUREG-1465.

These two levels of post-LOCA equipment qualification are discussed further below under items C and D.

Post LOCA Radiation Equipment Qualification

A. Equipment Groups

In the case of the LOCA safety related equipment needed for safe shutdown, mitigation, and post accident monitoring are divided into two functional groups:

<u>Group A:</u> Equipment needed for safe shutdown and post-accident mitigation, including RG 1.97 Type A variables.

<u>Group B:</u> Instrumentation needed to monitor plant status (including RG 1.97 Category 1 and 2 variables not included in Group A) during the accident as well as into the start of the recovery phase.

Components that fall under the Group A classification are qualified for component specific post accident durations which can range from accident initiation up to a maximum of 100 days. Three months is defined as the duration of the accident.

Components that fall under the Group B classification are qualified for component specific post accident durations which can range from accident initiation up to a maximum of 180 days. Six months is considered to be well into the recovery phase.

B. Qualification Time

The required qualification time for components (including margin requirements per RG 1.89, Rev 1) is developed as follows:

<u>Group A</u>

- For components needed in the short term (first 10 hours after the event), the qualification time is established based on a conservative estimate (consistent with the accident analyses) of when (and for how long) the component is required to function plus a margin of one hour.

- For components needed to operate intermittently or operate in the short term (but exceeding 10 hours), the qualification time is established based on a conservative estimate (consistent with the accident analyses) of when (and for how long) the component is required to be functional or until such time when an alternate method can be used to perform the function or when replacement components can be installed. Per RG 1.89, Rev 1, a 10% time margin is addressed. Add required operational periods

  In the event none of the above can be clearly established to be possible prior to 100 days following the DBA, the component is qualified to 100 days following the accident. The 100 days is assumed to include the 10% time margin requirements required by RG 1.89, Rev 1.

- For components needed to operate for the entire duration of the accident, the qualification time is 100 days or until such time when an alternate method can be used to perform the function or when replacement components can be installed. As discussed above, a 10% time margin is addressed.

Group B

- For components needed to operate for the entire duration of the accident as well as into the start of the recovery phase, the qualification time ranges up to 180 days, i.e.; until major recovery efforts are initiated and other monitoring techniques and/or devices appropriate for the specific event can be introduced. The qualification time for individual components are based on an evaluation of alternate methods that can be used to perform the function or when replacement components can be installed. Consideration is also given to the degree of deterioration, and impact on component function that is expected due to exposure during the extended period (i.e., beyond the 100 day duration of the accident), and whether compensatory techniques can be employed to maintain usability during that extended period (such as by addressing a further increase in instrument drift than addressed during the mitigation phase). As before, per RG 1.89, Rev 1, a 10% time margin is addressed for components required to be functional for time frames less than 180 days.

C. Qualification Level

The approach used in establishing the post LOCA radiation environments allows for the development of two functionally appropriate qualification levels. As discussed previously, these levels are based on the usage of radiation source terms (in the development of post-LOCA radiation environments) which are consistent with other system design bases. The components are segregated by required qualification level as follows:

Level 1 - Components Needed to Preclude or Limit Core Damage.

Level 2 - Components Needed to Maintain Containment Integrity (including in-vessel long-term core debris coolability).

D. Application of Qualification Level to Equipment and Instrumentation

Level 1

Environmental qualification for the ~~Safety Injection and~~ Emergency Feedwater System~~s~~ (i.e.; equipment needed to preclude or limit core damage) is based on the Draft NUREG-1465 100% Core Gap Release Source Term plus a margin as discussed below. Twenty percent of the core gap activity is assumed to be released instantaneously, as a puff, whereas the remaining 80% evolves in accordance with Draft NUREG 1465 over a period of 0.5 hrs. The puff release component of the model is intended to address fuel damage considerations from reactivity insertion events. The use of this source term is based on the following:

No change

- If the ~~SIS and~~ EFWS functions within its ~~their~~ design bases, ~~(both are designed against single failure and the SIS additionally meets Appendix K requirements)~~ there will be minimal core damage. STET

- In an "Arrested" Core melt Scenario, (i.e., if the effectiveness of the ~~above~~ safety systems is delayed resulting in the release of the gap plus some melted fuel):

  - The 100% gap qualification level (plus sufficient margin) will justify credit for restoration and operation of the ~~SIS and~~ EFWS for approximately three days to ~~arrest core damage and~~ help control reactor coolant system pressure.

  - Long term core cooling (i.e., beyond the three days) is guaranteed by ~~the cross connection provided between the containment spray~~ shutdown cooling pumps (Level 2 qualified equipment), and ~~the~~ safety injection ~~pump~~.

RG 1.97 Category 1 and 2 post-accident monitoring instrumentation that are qualified to Level 1 radiation environments are listed in Table 3.11-2. Summarized below is the basis for the use of this qualification level:

- Type A Variables - None

- Type B Variables:

  - Reactivity control monitoring instrumentation (neutron flux detectors) are primarily needed to show accomplishment of mitigation, and in this case establishing subcriticality which occurs almost immediately after the event. Additionally, in a Level 2 environment, the core geometry is lost due to core melt thus impacting long term neutron flux detection capability.

  - Core cooling monitoring instrumentation: reactor coolant hot/cold leg temperature can be established during a core melt scenario by the use of the unheated junction thermocouple (UHJTC) which is qualified to Level 2: reactor vessel coolant level and degrees of subcooling are not critical parameters for a core melt scenario.

- Type C Variables:

  - Fuel cladding monitoring instrumentation: Core exit temperature is useful in the early stages of heatup, up to the point where multiple fuel assembly pins have failed. Beyond this, potential increase in core exit temperature in a core melt scenario can be established by noting increase in containment temperature (containment temperature monitors are qualified to Level 2); radioactivity in circulating primary coolant is intended to detect pin failures. This function is completed very early in the accident and survivability through the entire core melt scenario is not considered necessary. RCS pressure boundary monitoring instrumentation: though listed as a Category 3 instrumentation, qualification of the containment area monitors is required since credit is taken for their operation (and subsequent containment isolation) in the site boundary and control room dose analysis following a CEA ejection accident (anticipated fuel failure is 6.8%).

- Type D Variables:

  - ~~Safety Injection System monitoring instrumentation: As discussed earlier in this Section, the entire Safety Injection System is qualified to Level 1 which includes the accumulator tank level/pressure, accumulator isolation valve position and safety injection flow.~~

  - Primary Coolant System monitoring instrumentation: Primary System safety-valve position indication, pressurizer level and heater status is not necessary for the duration of the core melt scenario since its monitoring function is completed early in the accident when the Safety Depressurization System is initiated.

  - Secondary System monitoring instrumentation: SG level/pressure and safety/relief valve position is not needed for the duration of the core melt scenario since its monitoring function is completed early in the accident when the Safety Depressurization System is initiated (i.e., long term RCS pressure control will not be achieved by steam dump via the SG).

  - Emergency Feedwater System monitoring instrumentation: As discussed earlier in this section, the entire Emergency Feedwater System is qualified to Level 1 which includes the emergency feedwater flow and storage tank level.

- Containment sump water temperature instrumentation: Potential increase in sump water temperature in a core melt scenario can be established by noting increase in containment temperature (containment temperature monitors are qualified to Level 2).

• Type E Variables:

- Airborne radioactivity materials released from the plant: The primary purpose of the main steam line radiation monitors is radiation detection following a steam generator tube rupture. These detectors do not need qualification for the duration of the core melt scenario since long term RCS pressure control is achieved by the Safety Depressurization System rather than by steam releases via the SG. long-term

Level 2 (including in-vessel core debris cooling)

Environmental qualification for the components needed to maintain containment integrity is based on Draft NUREG 1465 Gap Plus Early In-vessel Release Source Term to satisfy the "substantial" core melt postulated by 10CFR100 which presupposes that emergency core cooling has failed initially.

RG 1.97 Category 1 and 2 post-accident monitoring instrumentation not addressed for Level 1 qualification above will be qualified to appropriate radiation environments bounded by Level 2 qualification requirements.

Table 3.11-2 summarizes the assigned qualification level (i.e., 1 or 2) for Group A and Group B components.

REFERENCES FOR SECTION 3.11

1. "Qualification of Combustion Engineering Class 1E Instrumentation," CENPD-255-A Revision 3, Combustion Engineering, Inc., October 1985.

2. "Qualification Guidelines for Instrumentation and Controls Equipment for Nuplex 80+," NPX80-IC-QG790-00.

3. "Accident Source Terms for Light Water Nuclear Power Plants," Draft NUREG-1465, June 1992.

4. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Immediately Following an Accident," Revision 3, May 1983.

5. Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," Revision 1, June 1984.

TABLE 3.11-1

VENTILATION AREAS

Area Name	Safety-Related Yes	Safety-Related No	Area Temp Alarm Control Room	Exhaust Duct Temp. Alarm Control Room
A. Control Building				
1. Control Room and Adj. Offices	X		X	X
2. Computer Room	X		X	
3. Vital Instrument & Equipment	X		X	
4. Essential Switchgear Room	X		X	
5. CEDM Control Room		X	X	
6. Essential Battery Room	X		X	X
7. Remote Shutdown Panel	X		X	
B. Subsphere				X
1. Turbine Driven FWP Room	X		X	
2. Motor Driven FWP Room	X		X	
3. SCS Heat-Exchanger Room	X		X	
4. SI Pump Room	X		X	
5. CS Pump Room	X		X	
6. Containment Spray Heat-X Room	X		X	
7. Fuel Pool Heat Exchange Room	X		X	
8. Fuel Pool Cooling Pump Room	X		X	
9. Penetration Room	X		X	
C. Nuclear Annex				
1. Essential Chiller and Pump	X		X	
2. Component Cooling Water Pump Room	X		X	
D. Diesel Generator Building	X		X	
E. Fuel Build. Exhaust Filter Train	X		X	X
F. Containment		X	X	

TABLE 3.11-2

(Sheet 1 of 2)

SUMMARY OF ASSIGNED
RADIOLOGICAL EQUIPMENT QUALIFICATION LEVEL

RADIATION QUALIFICATION LEVEL	EQUIPMENT GROUP A	EQUIPMENT GROUP B
LEVEL 1*	Safety Injection System Emergency Feedwater System	SEE LIST 1
LEVEL 2	Containment Pressure Boundary (Including Containment Isolation) Safety Depressurization System Shutdown Cooling System Containment Spray System Combustible Gas Control** Component Cooling Water System*** Essential Chilled Water Systems*** Equipment and Floor Drains*** Subsphere and Annulus Ventilation Systems	SEE LIST 2

LIST 1	LIST 1 (Continued)
Primary Safety Valve Position	IRWST Temperature
Reactor Coolant Temp (hot/cold)	Emergency Feedwater Flow
Reactor Coolant Radiation Level	Emergency Feedwater Storage Tank Level
Reactor Vessel Coolant Level	Safety Injection Flow
SG Pressure	Safety Injection Tank Level
SG Level	Safety Injection Tank Pressure
SG Safety Valve/ADV Position	Safety Injection Tank Isolation Valve Position
Pressurizer Level	Main Steam Line Effluent Radiation Level
Pressurizer Heater Status	Containment Area Radiation (Low Range)
Degree of Subcooling	
Neutron Flux	
Core Exit Temperature	

add to "List 2" on next page.

* Plus Margin for Transition to Level 2

** Equipment Qualification per Regulatory Guide 1.7

*** Portions Supporting Containment Spray/Shutdown Cooling

TABLE 3.11-2

(Sheet 2 of 2)

SUMMARY OF ASSIGNED
RADIOLOGICAL EQUIPMENT QUALIFICATION LEVEL

RADIATION QUALIFICATION LEVEL	EQUIPMENT GROUP A	EQUIPMENT GROUP B
LIST 2	LIST 2 (Continued)	
RCS Pressure	Shutdown Cooling HX Outlet Temperature	
SDS Valve Position	IRWST Level	
SDS Pressure	Containment Isolation Valve Position	
SDS Temperature	Containment Area Radiation Monitor (high range)	
UHJTC	Unit Vent Post Accident Concentration	
Containment Pressure	Unit Vent Flow	
Containment Temperature	Control Room Ventilation Damper Position	
Containment Hydrogen Concentration****	Status of Standby Power & other Safety Related Energy Sources	
Containment Spray Flow	Component Cooling Water flow & temperature to ESF System	
Shutdown Cooling Flow		

Add Safety Injection List
from Sheet 1 of this table.

**** Equipment Qualification per Regulatory Guide 1.7

TABLE 3.11A-1 (Cont'd)

(Sheet 8 of 8)

ENVIRONMENTAL DATA

Station Service Water Pump Structure/Component Cooling Water
Heat Exchanger Structure Environmental Data
Environmental Category O
(LOCA/MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	125, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads[1]	N/A
Chemical Spray	N/A

NOTES:

1. Accident condition gamma radiation dose includes the normal external gamma dose plus that external dose due to the limiting DBA since these are total integrated dose values. The component design dose is the sum of internal (if applicable) plus external radiation doses.

2. Environment as used in this Table is defined as those conditions surrounding equipment. Equipment specifications take into consideration both the environment and those process conditions internal to the equipment.

3. Outside the biological shield.

4. The post-LOCA radiation environment in this region will vary depending on whether or not emergency core cooling operates within its design basis. If emergency core cooling operates as designed, there will be little core damage and a conservative estimate of the radiological release would be 100% of the core gap activity. If emergency core cooling is assumed to fail in the short-term but is restored to operation resulting in an "arrested core damage" scenario (to be consistent with the "substantial" core melt accident postulated to satisfy 10 CFR 100), the radiological release is assumed to be 100% of the core gap activity as well as the early in-vessel core release as discussed in Draft NUREG-1465. Table 3.11A-1 assumes an arrested core melt scenario integrated over six months and is intended to provide an upper bound radiation environment for the region.

5. Post-LOCA radiation environment in the Hydrogen Recombiner Cubicle is based on the guidance provided in Regulatory Guide 1.7.

EFFECTIVE PAGE LISTING

APPENDIX 3.11A

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3.11A-5A	U
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APPENDIX 3.11A

ENVIRONMENTAL CONDITIONS AND TEST PROFILES

FOR

STRUCTURES AND COMPONENTS

LIST OF FIGURES

APPENDIX 3.11A

APPENDIX 3.11A

ENVIRONMENTAL CONDITIONS AND TEST PROFILES

FOR

STRUCTURES AND COMPONENTS

1.0 PURPOSE

The purpose of Appendix 3.11A is to present typical environmental design data for normal conditions and "worst case" environmental design data for accident conditions. These data were developed from the experience of operating plants modified for the unique design of System 80+. The "worst case" post-LOCA radiation environments are based on the six month integrated gap and early in-vessel releases discussed in Draft NUREG-1465.

2.0 DISCUSSION

The tables and figures in this appendix show categories which are associated with particular regions of the plant, plus either normal or accident conditions. Specifying a category for a piece of equipment fixes the worst case environment at its location. The equipment, however, is qualified only to the environmental radiation exposure received during normal operation and for the duration of time after the worst case DBA for which it is required to be functional. The typical test profiles shown include test margins required by 10 CFR 50.49. The "inside cabinet" test profiles (Figures 3.11A-8 and 3.11A-10) include a temperature margin which accounts for heating effects inside the cabinet. Figures 3.11A-9 and 3.11A-10 include allowance for loss of ventilation in certain regions of the Nuclear Annex/Subsphere. Generic testing of equipment may, and usually does, exceed the conditions shown in the test profiles.

Add

both during and

TABLE 3.11A-1

(Sheet 1 of 8)

ENVIRONMENTAL DATA

Containment Vessel Environmental Data
Environmental Category A-1
(LOCA)

Environmental Parameters	Range and Duration
Temperature, °F	Figure 3.11A-1A
Pressure, psig	Figure 3.11A-1B
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads Plus LOCA[1,3,4]	$< 4.3 \times 10^7$ Gamma $< 3.5 \times 10^8$ Beta
Chemical Spray	4,400 ppm Boron as H_3BO_3 pH of 7.0-8.5 after 4 hours using Trisodium phosphate

Containment Vessel Environmental Data
Environmental Category A-2
(MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	Figure 3.11A-3 0-30 min. Figure 3.11A-1A after 30 min.
Pressure, psig	Figure 3.11A-1B
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads[1,3]	$< 3.1 \times 10^6$ Gamma
Chemical Spray	4,400 ppm Boron as H_3BO_3 pH of 7.0-8.5 after 4 hours using Trisodium phosphate

TABLE 3.11A-1 (Cont'd)

(Sheet 2 of 8)

ENVIRONMENTAL DATA

Annulus Environmental Data
Environmental Category A-3
(Post DBA)

Environmental Parameters	Range and Duration
Temperature, °F	Figure 3.11A-2
Pressure, psig	atmospheric, continuous
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads Plus LOCA[4]	3×10^5 Gamma 4×10^5 Beta
Chemical Spray	N/A

Containment Vessel Environmental Data
Environmental Category B
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	60-110, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, continuous
Radiation, 60 Yr. TID Rads[3]	$< 3 \times 10^6$ Gamma
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 3 of 8)

ENVIRONMENTAL DATA

Nuclear Annex/Subsphere Environmental Data
Environmental Category C
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	55-104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, continuous
Radiation, 60 Yr. TID Rads	$< 1 \times 10^3$ Gamma Hydrogen Recombiner Rms. $< 1.5 \times 10^3$ Gamma Reactor Bldg. Subsphere $< 1 \times 10^3$ Gamma Component Cooling Pump Rms., EFW Pump Rms., Essential Chillers Rms.
Chemical Spray	N/A

Nuclear Annex/Subsphere Environmental Data
Environmental Category C1
(Abnormal - Loss of HVAC)

Environmental Parameters	Range and Duration
Temperature, °F	55-122
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, continuous
Radiation, 60 Yr. TID Rads	$< 1 \times 10^3$ Gamma, Hydrogen Recombiner Rms. $< 1.5 \times 10^6$ Gamma, Reactor Bldg. Subsphere $< 1 \times 10^3$ Gamma, Component Cooling Pump Rms., EFW Pump Rms. and Essential Chillers Rms.
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 4 of 8)

ENVIRONMENTAL DATA

Nuclear Annex/Subsphere Environmental Data
Environmental Category D
(LOCA/MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	55-104
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, Limited to 8 hours outside normal range of Category C.
Radiation, 60 Yr. TID Rads[1] plus LOCA[4]	$< 3 \times 10^5$ Gamma Hydrogen[5] Recombiner Rms. $< 2.1 \times 10^7$ Gamma Reactor Bldg. Subsphere (assumes streaming from inside containment) $< 1 \times 10^3$ Gamma Component Cooling Pump Rms., EFW Pump Rms., Essential Chillers Rms. $< 5 \times 10^6$ Gamma (ESF/Annulus Bld. filler cubicles)
Chemical Spray	N/A

Mechanical Equipment Room
Environmental Category E
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	$\leq 10^3$ Gamma
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 5 of 8)

ENVIRONMENTAL DATA

Spent Fuel Pool Area Environmental Data
Environmental Category F
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	40-104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	$\leq 10^3$ Gamma
Chemical Spray	N/A

Spent Fuel Pool Area Environmental Data
Environmental Category G
(LOCA/MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	40-104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads[1]	$\leq 10^3$ Gamma
Chemical Spray	N/A

Emergency Diesel Generator Areas Environmental Data
Environmental Category H
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	40-120, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rad	$< 10^3$ Gamma
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 6 of 8)

ENVIRONMENTAL DATA
Emergency Diesel Generator Areas Environmental Data
Environmental Category I
(LOCA/MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	125, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads[1]	$< 10^3$ Gamma
Chemical Spray	N/A

Control Room Environmental Data Environmental
Category J
(Normal/DBA)

Environmental Parameters	Range and Duration
Temperature, °F	73-78, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-60, continuous
Radiation, 60 Yr. TID Rads	$\leq 10^3$ Gamma
Chemical Spray	N/A

Control Area Battery Rooms Environmental Data
Environmental Category P
(Normal)

Temperature, °F	77, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	$\leq 10^3$ Gamma
Chemical Spray	N/A

Other Control Areas Environmental Data
Environmental Category K
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	85, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	$\leq 10^3$ Gamma
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 7 of 8)

ENVIRONMENTAL DATA

Main Steam Valve House Environmental Data
Environmental Category L
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	40-115, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	< 10^3 Gamma
Chemical Spray	N/A

Main Steam Valve House Environmental Data
Environmental Category M
(MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	300 max
Pressure, psig	10
Relative Humidity, %	100, continuous
Radiation, 60 Yr. TID Rads[1]	< 10^3 Gamma
Chemical Spray	N/A

Station Service Water Pump Structure/Component Cooling Water
Heat Exchanger Structure Environmental Data
Environmental Category N
(Normal)

Environmental Parameters	Range and Duration
Temperature, °F	40-104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads	N/A
Chemical Spray	N/A

TABLE 3.11A-1 (Cont'd)

(Sheet 8 of 8)

ENVIRONMENTAL DATA

Station Service Water Pump Structure/Component Cooling Water
Heat Exchanger Structure Environmental Data
Environmental Category O
(LOCA/MSLB)

Environmental Parameters	Range and Duration
Temperature, °F	125, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-100, continuous
Radiation, 60 Yr. TID Rads[1]	N/A
Chemical Spray	N/A

NOTES:

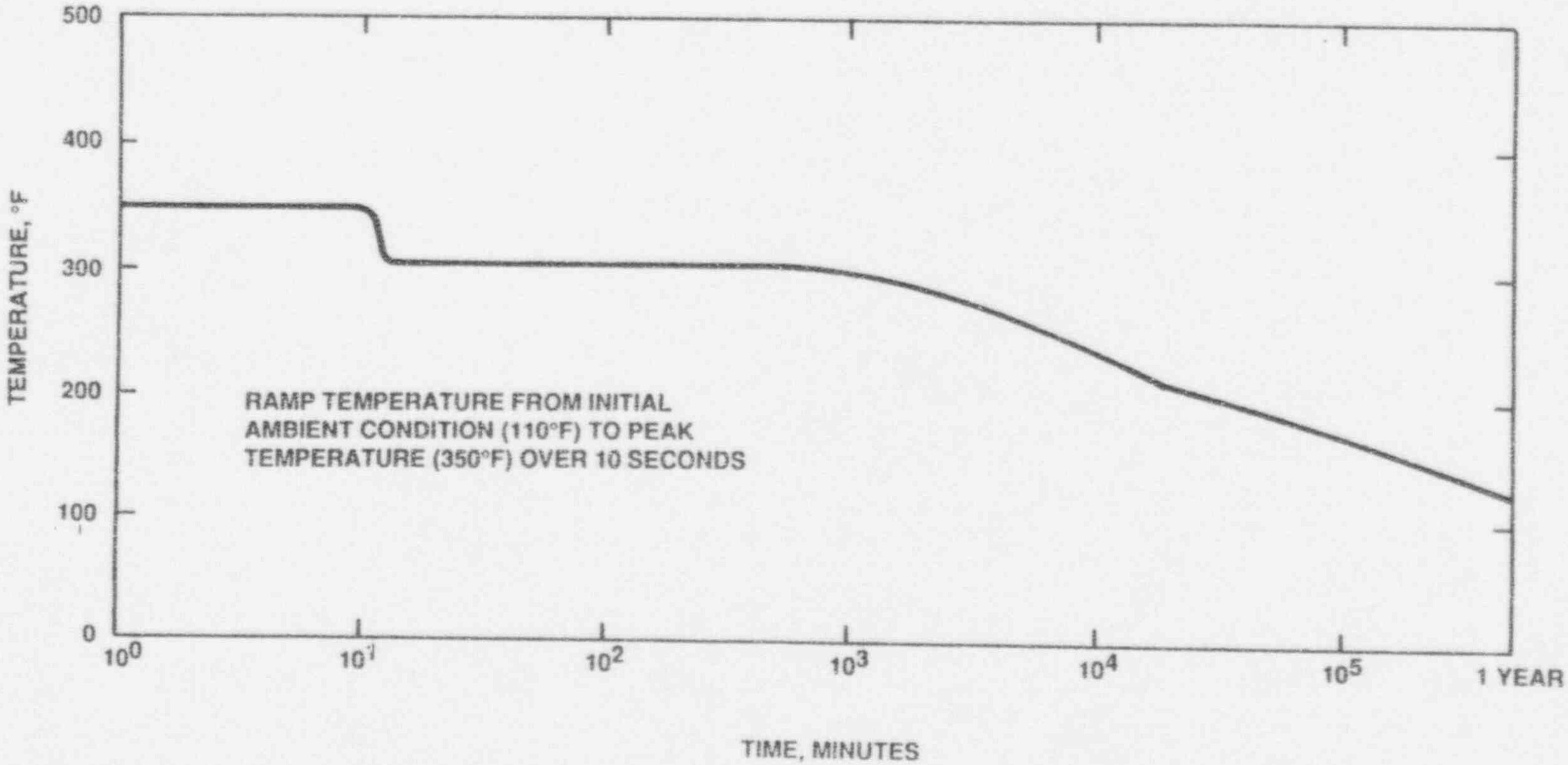
1. Accident condition gamma radiation dose includes the normal external gamma dose plus that external dose due to the limiting DBA since these are total integrated dose values. The component design dose is the sum of internal (if applicable) plus external radiation doses.

2. Environment as used in this Table is defined as those conditions surrounding equipment. Equipment specifications take into consideration both the environment and those process conditions internal to the equipment.

3. Outside the biological shield.

4. The post-LOCA radiation environment in this region will vary depending on whether or not emergency core cooling operates within its design basis. If emergency core cooling operates as designed, there will be little core damage and a conservative estimate of the radiological release would be 100% of the core gap activity. If emergency core cooling is assumed to fail in the short-term but is restored to operation resulting in an "arrested core damage" scenario (to be consistent with the "substantial" core melt accident postulated to satisfy 10 CFR 100), the radiological release is assumed to be 100% of the core gap activity as well as the early in-vessel core release as discussed in Draft NUREG-1465. Table 3.11A-1 assumes an arrested core melt scenario integrated over six months and is intended to provide an upper bound radiation environment for the region.

5. Post-LOCA radiation environment in the Hydrogen Recombiner Cubicle is based on the guidance provided in Regulatory Guide 1.7.

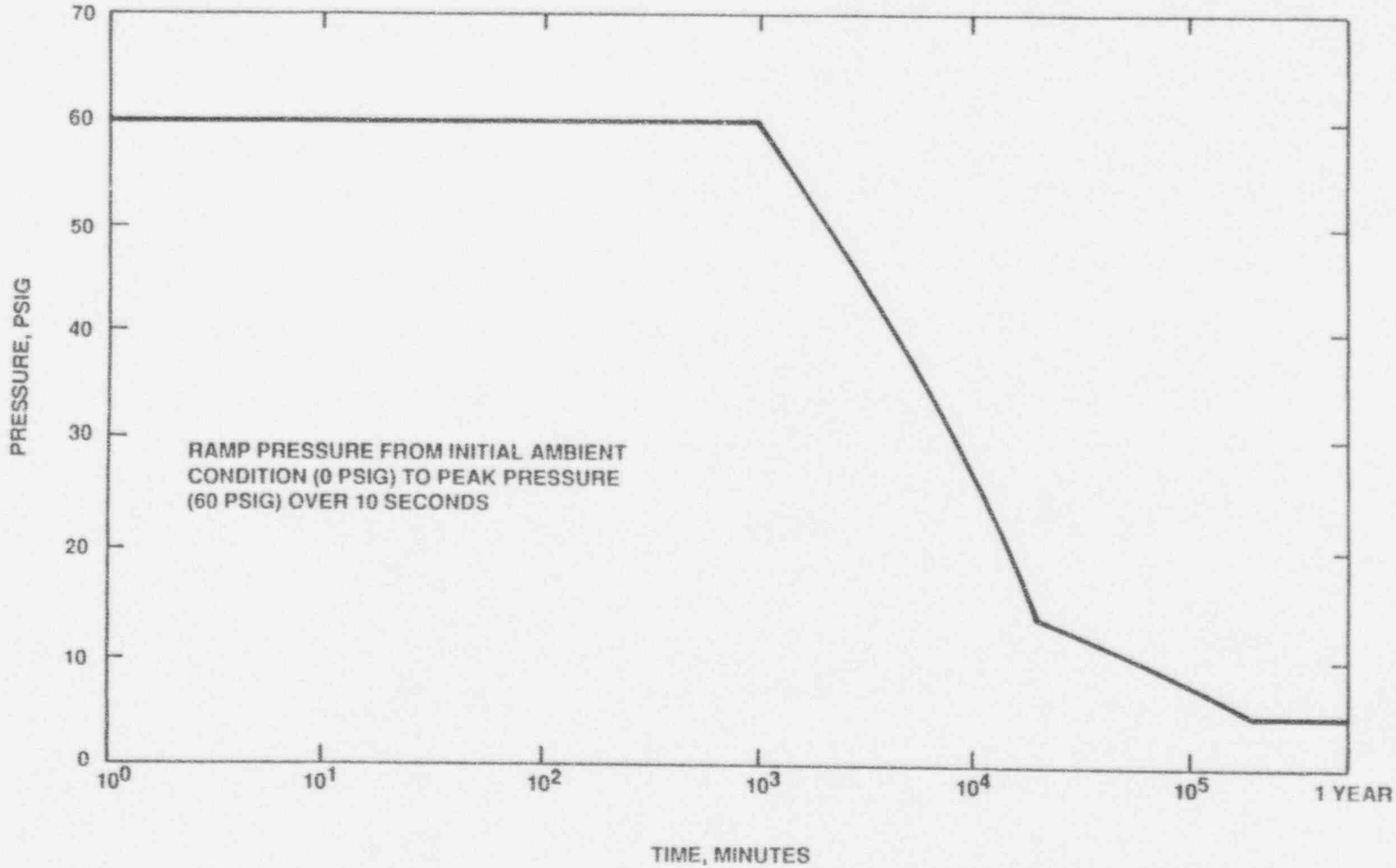


SYSTEM 80+™

DESIGN BASIS CONTAINMENT ATMOSPHERE TEMPERATURE CONDITION FOLLOWING LOCA

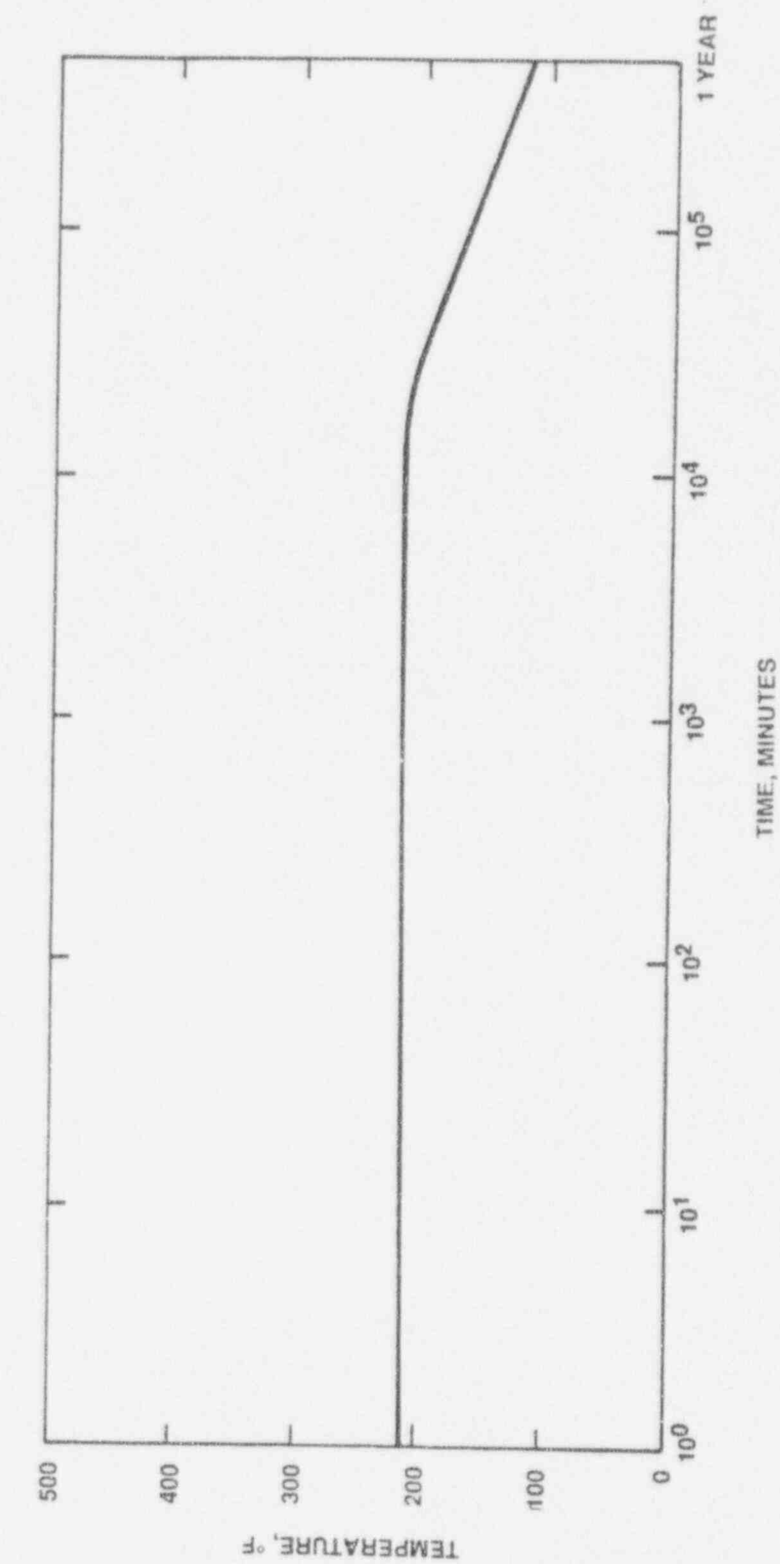
Figure 3.11A - 1A

SYSTEM 80+™



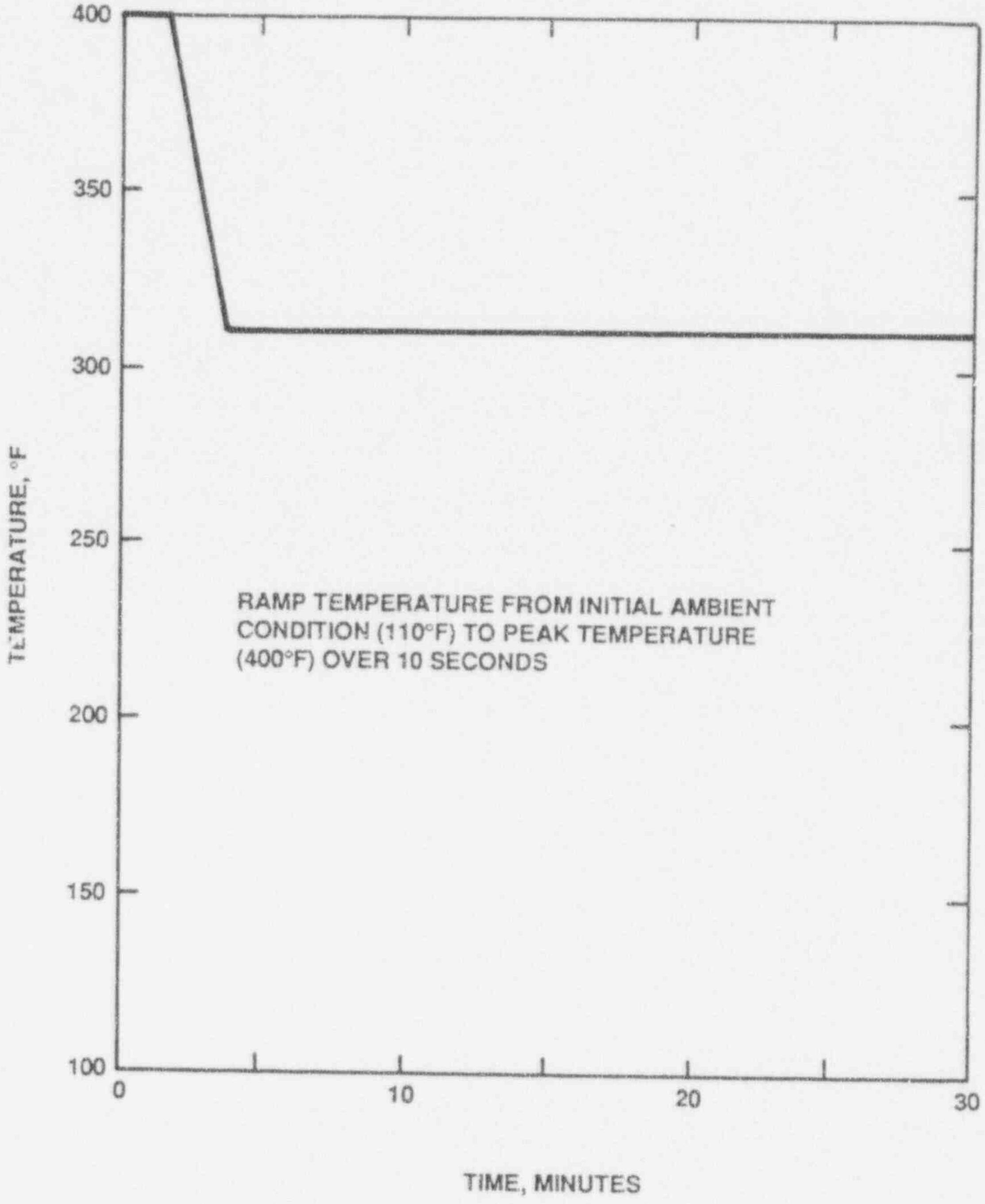
DESIGN BASIS CONTAINMENT ATMOSPHERE PRESSURE CONDITION FOLLOWING LOCA

Figure 3.11A - 1B

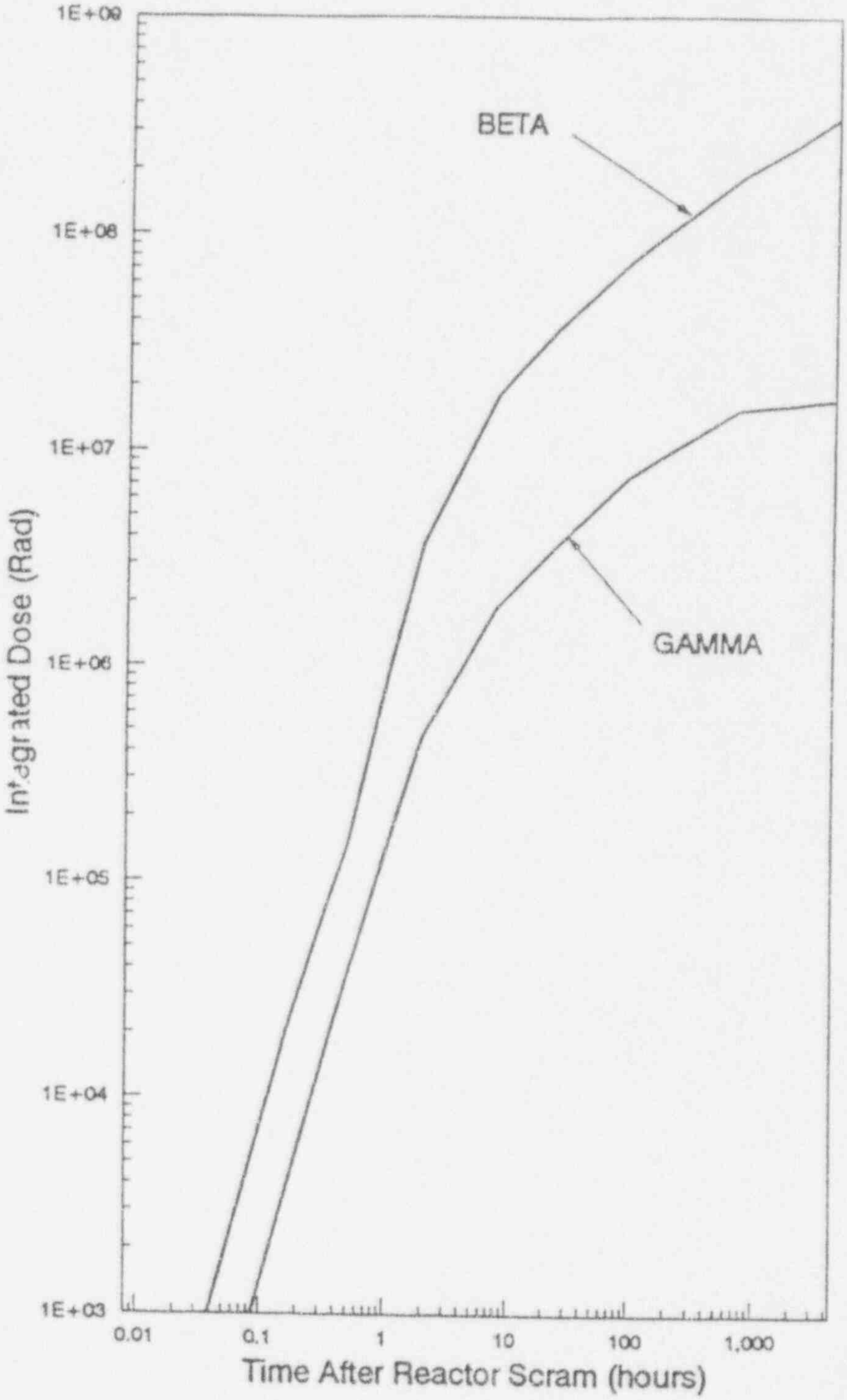


Amendment U - 12/31/93

SYSTEM 80+™	DESIGN BASIS CONTAINMENT ATMOSPHERE TEMPERATURE CONDITION FOLLOWING LOCA/MSLB	Figure 3.11A - 2

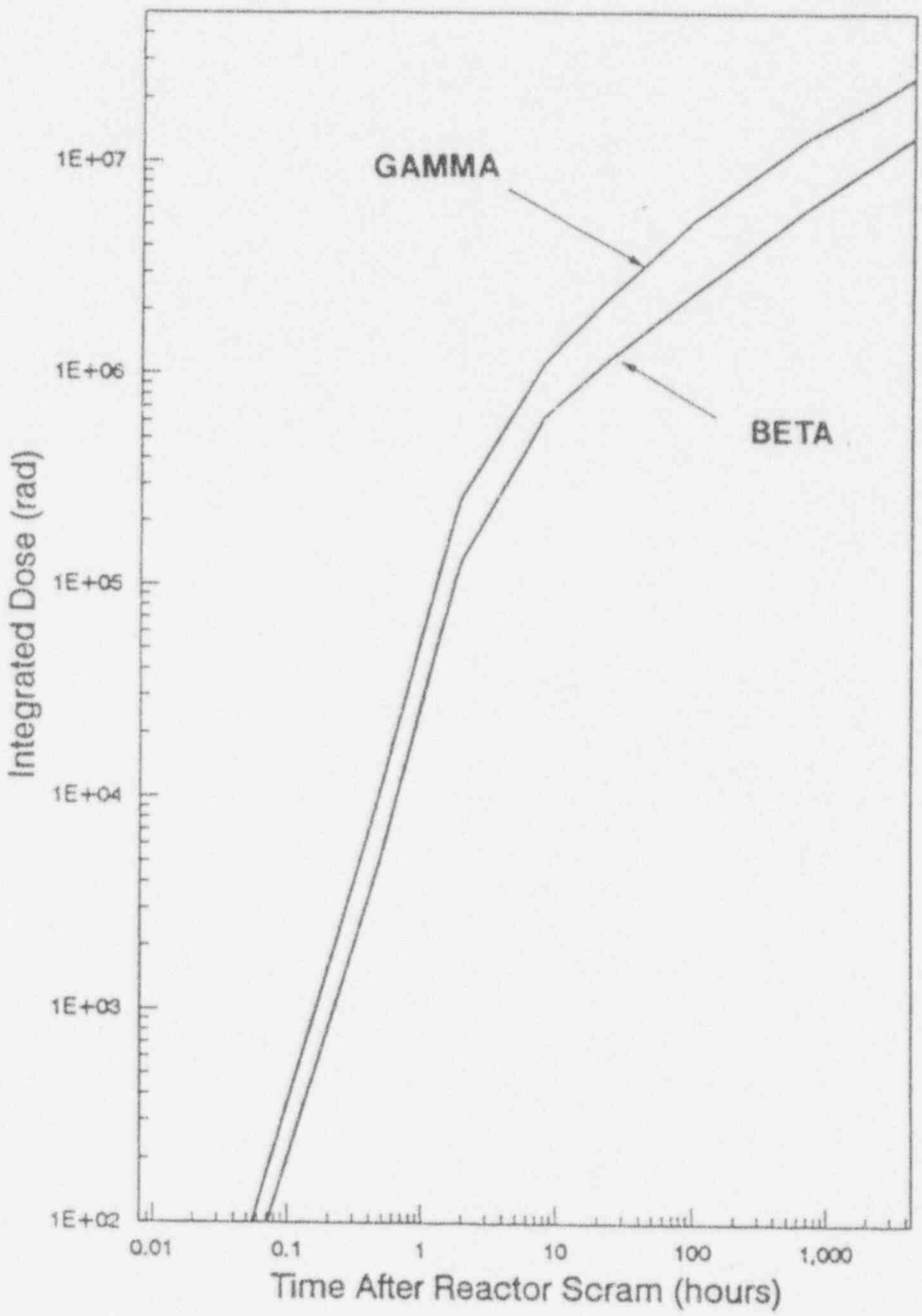


SYSTEM 80+™	DESIGN BASIS CONTAINMENT ATMOSPHERE TEMPERATURE CONDITION FOLLOWING MSLB	Figure 3.11A - 3



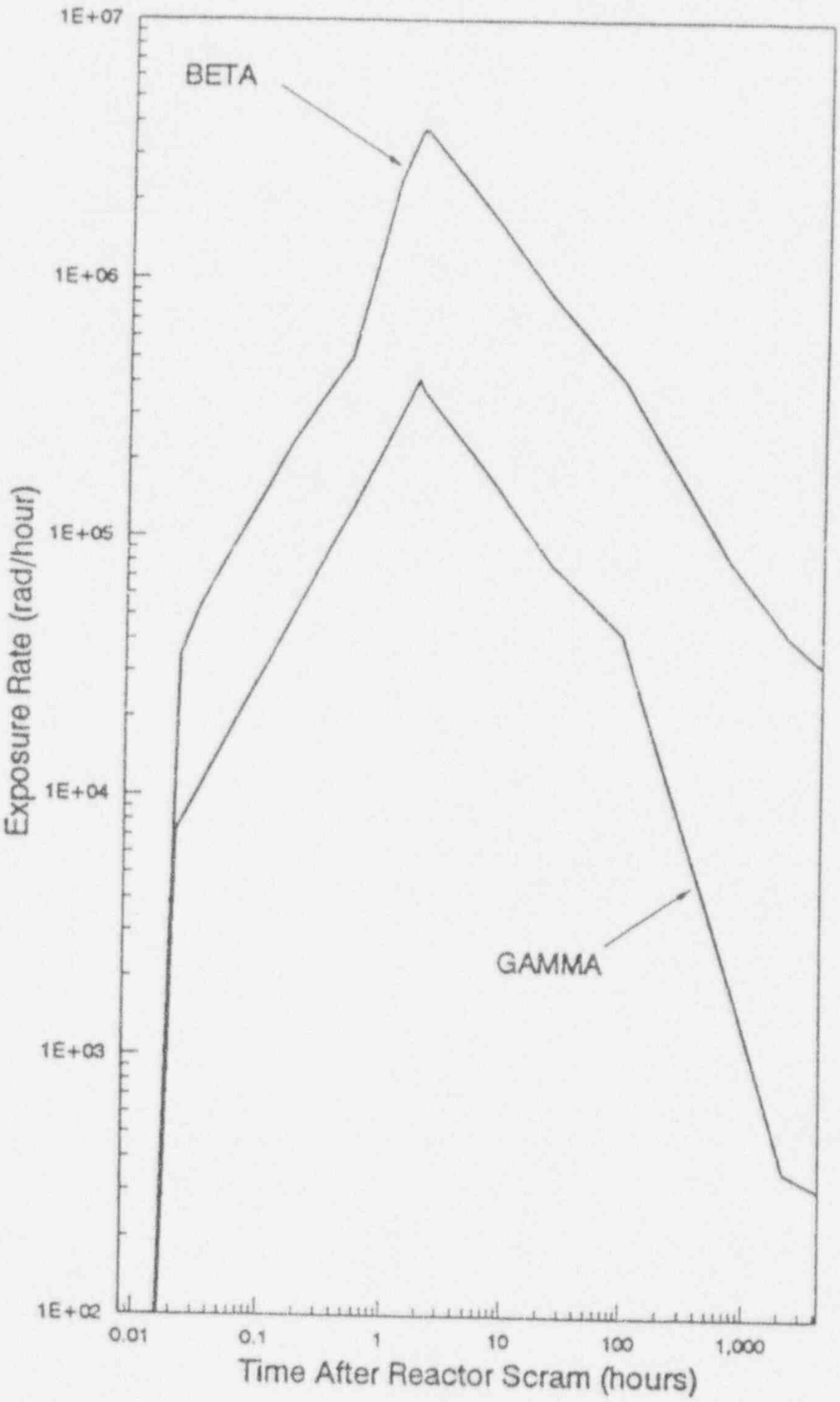
Amendment U - 12/31/93

SYSTEM 80+™	WORST CASE (LEVEL 2) INTEGRATED CONTAINMENT ATMOSPHERE RADIATION DOSE FOLLOWING LOCA	Figure 3.11A - 4A



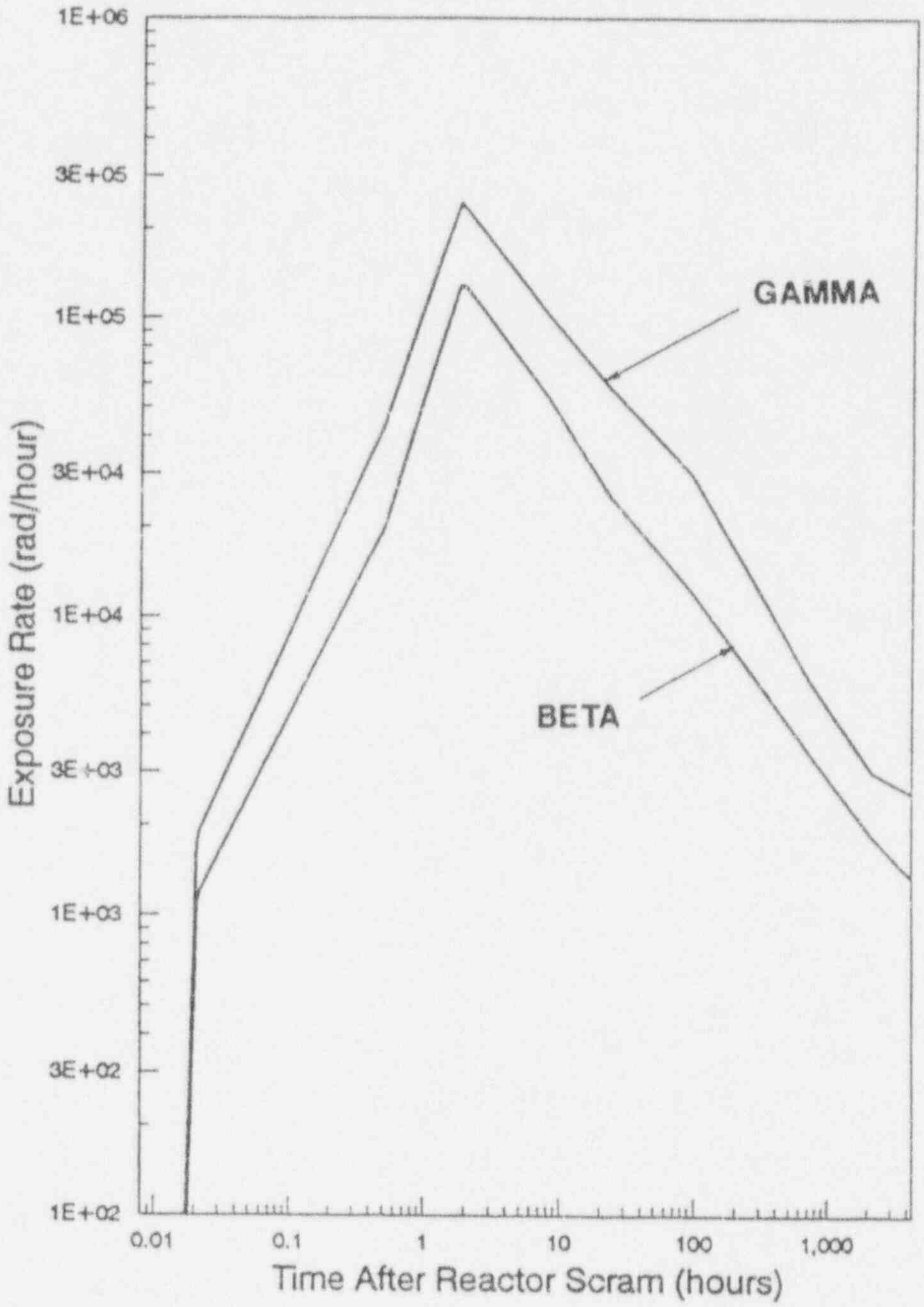
Amendment U - 12/31/93

SYSTEM 80+™	WORST CASE (LEVEL 2) INTEGRATED CONTAINMENT IRWST RADIATION DOSE FOLLOWING LOCA	Figure 3.11A - 4B



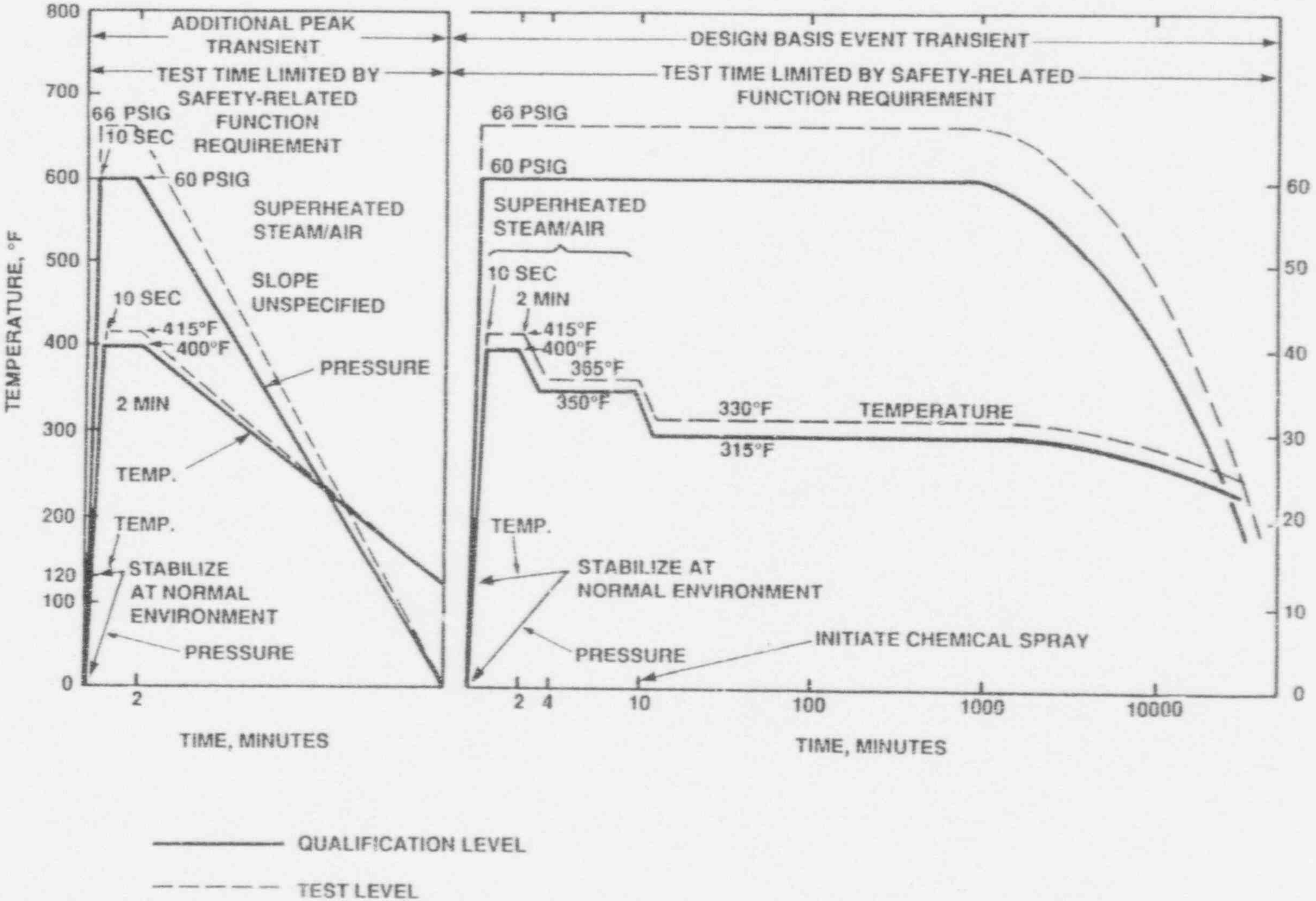
Amendment U - 12/31/93

SYSTEM 80+™	WORST CASE (LEVEL 2) CONTAINMENT ATMOSPHERE EXPOSURE RATES FOLLOWING LOCA	Figure 3.11A - 5A



Amendment U - 12/31/93

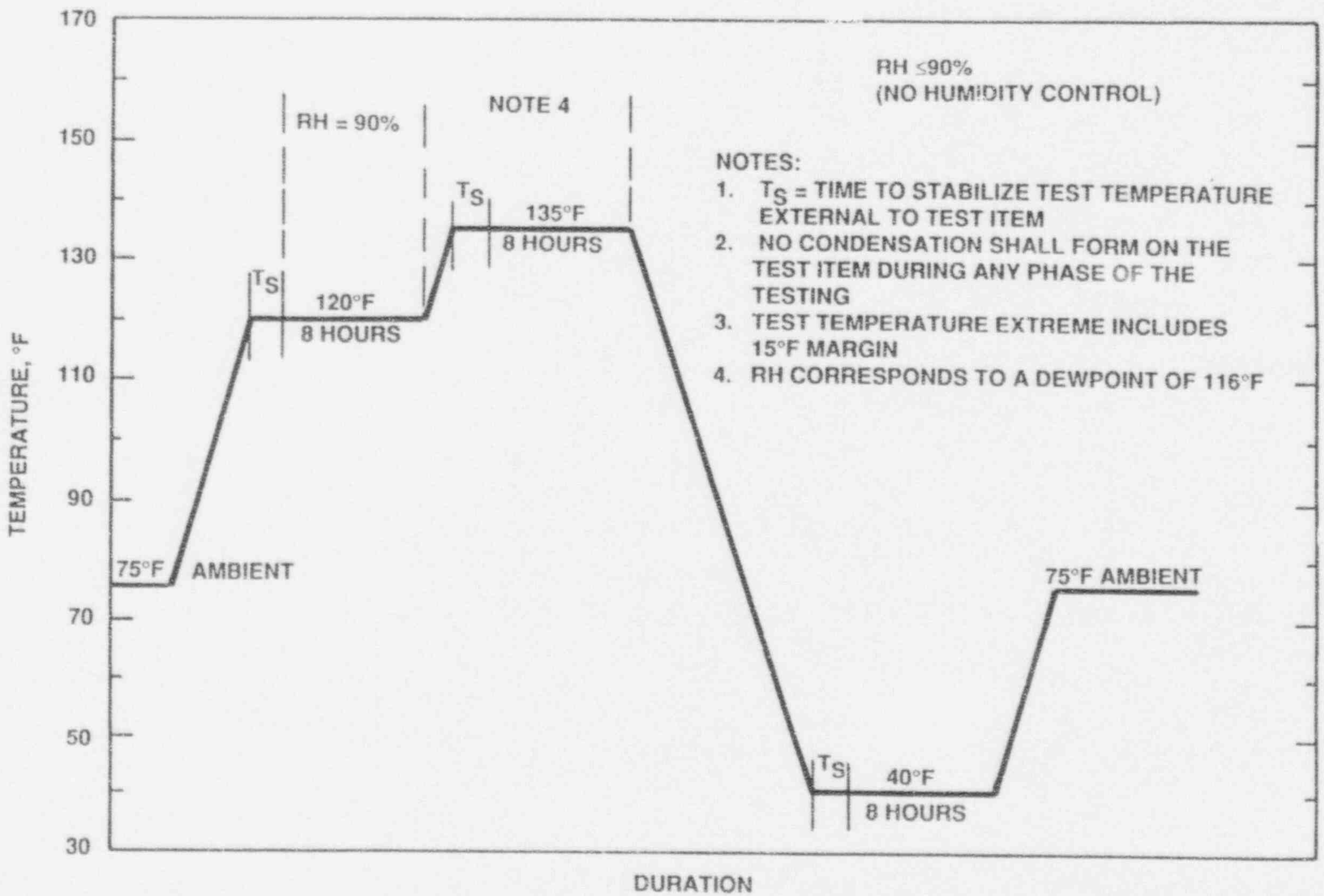
SYSTEM 80+™	WORST CASE (LEVEL 2) CONTAINMENT IRWST EXPOSURE RATES FOLLOWING LOCA	Figure 3.11A - 5B



SYSTEM 80+™

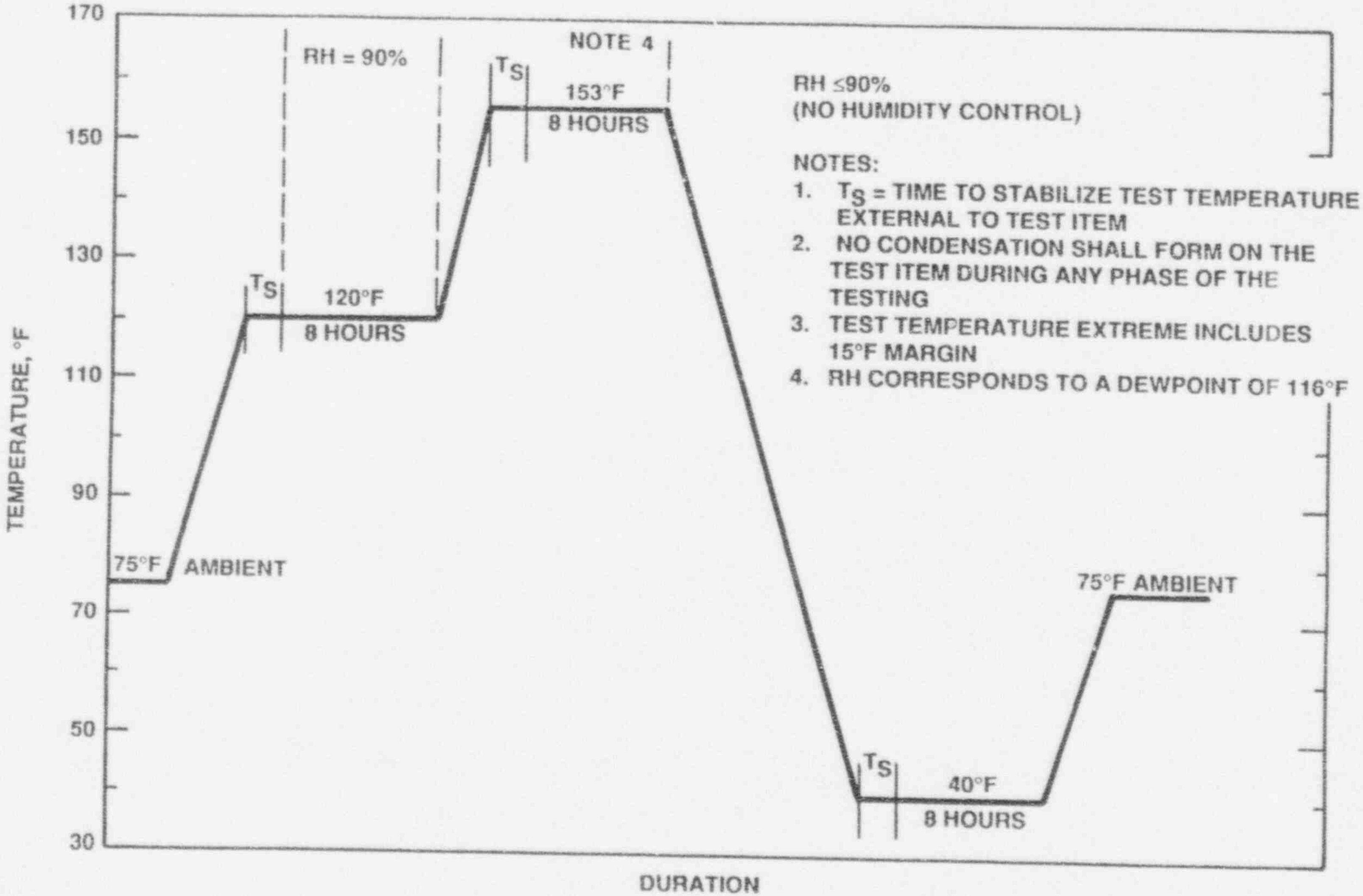
DESIGN BASIS CONTAINMENT BUILDING ENVIRONMENTAL TEST PROFILE FOR CATEGORY "A-1" AND "A-2"

Figure 3.11A - 6



DESIGN BASIS ENVIRONMENTAL TEST PROFILE FOR NUCLEAR ANNEX, CATEGORY D, LOCA/MSLB

Figure 3.11A - 7

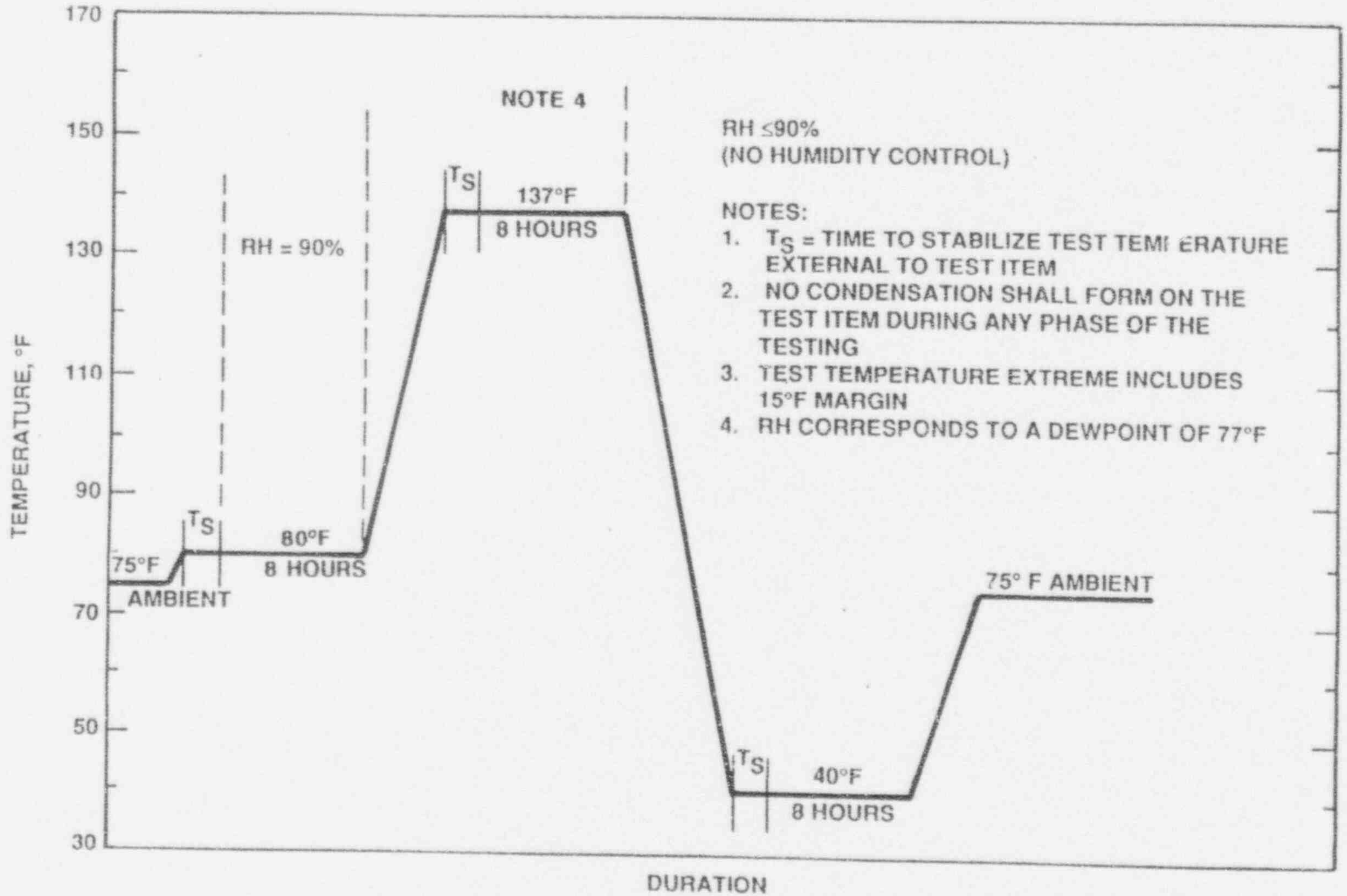


System 80+™

DESIGN BASIS INSIDE CABINET ENVIRONMENTAL TEST PROFILE FOR NUCLEAR ANNEX/SUBSPHERE, CATEGORY D

Figure 3.11A - 8

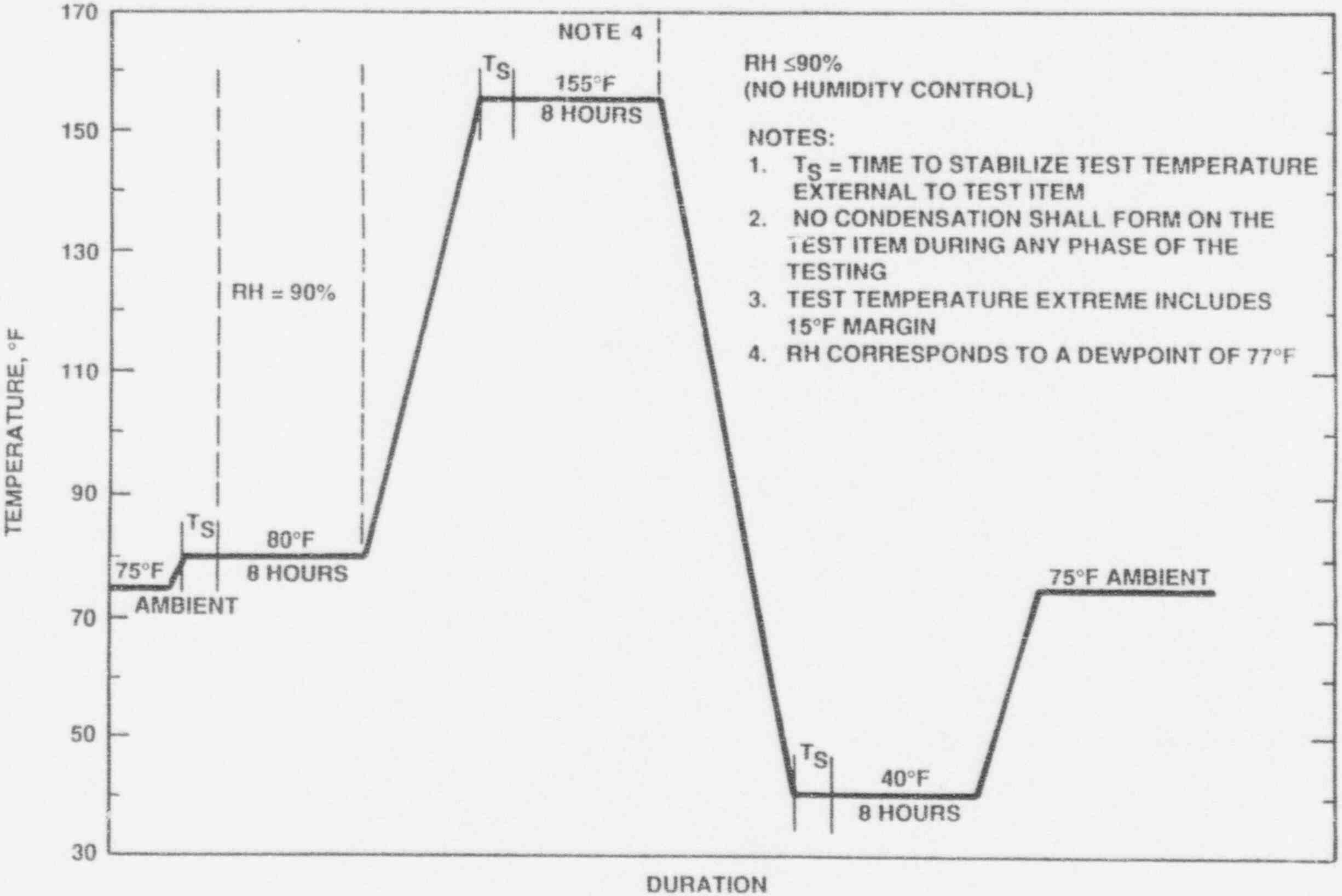
SYSTEM 80+™



Amendment U - 12/31/93

DESIGN BASIS OUTSIDE CABINET ENVIRONMENTAL TEST PROFILE FOR NUCLEAR ANNEX/SUBSPHERE, CATEGORY "C1" ENVIRONMENTAL CONDITIONS

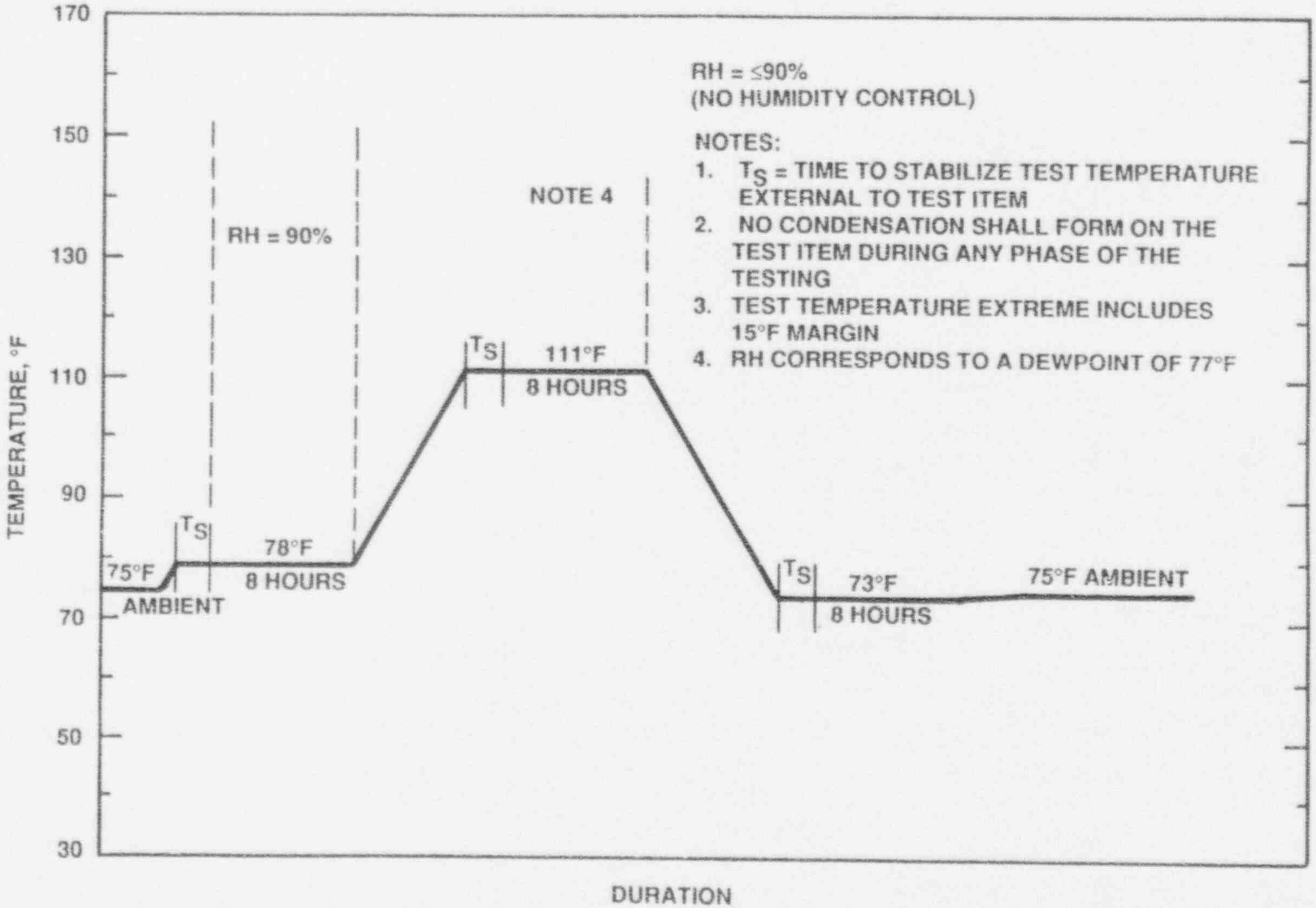
Figure 3.11A - 9



SYSTEM 80+™

DESIGN BASIS INSIDE CABINET ENVIRONMENTAL TEST PROFILE FOR NUCLEAR ANNEX/SUBSPHERE CATEGORY "C1" ENVIRONMENTAL CONDITIONS

Figure 3.11A - 10



DESIGN BASIS INSIDE CABINET ENVIRONMENTAL TEST PROFILE FOR CATEGORY "J" ENVIRONMENTAL CONDITIONS

Figure 3.11A - 11

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3.11B-1 (Sheet 3)	U
3.11B-1 (Sheet 4)	U
3.11B-1 (Sheet 5)	U
3.11B-1 (Sheet 6)	U
3.11B-1 (Sheet 7)	U
3.11B-1 (Sheet 8)	U
3.11B-1 (Sheet 9)	U
3.11B-1 (Sheet 10)	U
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3.11B-2 (Sheet 1)	U
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APPENDIX 3.11B

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APPENDIX 3.11B

APPENDIX 3.11B

IDENTIFICATION, LOCATION AND ENVIRONMENTAL CONDITIONS OF EQUIPMENT

ABSTRACT

Appendix 3.11B presents in tabular form the identification of equipment which will be environmentally qualified, the plant location of that equipment, and the worst-case environmental conditions associated with the region in which the equipment is located. The location within the SAR where the equipment is discussed is also listed.

TABLE 3.11B-1

(Sheet 1 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Safety Injection	Continuous	Continuous	D, Subsphere	Safety Injection Pump and Motor	6.3.2.2.3
Shutdown Cooling	Continuous	Continuous	D, Subsphere	Shutdown Cooling Pump and Motor	5.4.7.2.2
Containment Spray	Continuous	Continuous	D, Subsphere	Containment Spray Pump and Motor	6.5.2.2.1
Emergency Feedwater	Continuous	Continuous	D, Subsphere	Motor-Driven Emergency Feedwater Pump 1 and Motor	10.4.9.2.2.1
Emergency Feedwater	Continuous	Continuous	D, Subsphere	Motor-Driven Emergency Feedwater Pump 2 and Motor	10.4.9.2.2.1
Emergency Feedwater	Continuous	Continuous	D, Subsphere	Steam-Driven Emergency Feedwater Pump 1 and Motor	10.4.9.2.2.2
Emergency Feedwater	Continuous	Continuous	D, Subsphere	Steam-Driven Emergency Feedwater Pump 2 and Motor	10.4.9.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 332, Globe Valve and Actuator, Hot Leg Check Valve Leakage Isolation	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 322, Globe Valve and Actuator, Hot Leg Check Valve Leakage Isolation	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 670, Globe Valve and Actuator, IRWST Return Isolation, CIV	6.3.2.2.5

Refer to last page of this table for footnote definitions.

TABLE 3.11B-1 (Cont'd)

(Sheet 2 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 608, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 606, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 607, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 605, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 641, Globe Valve and Actuator, SIT Fill/Drain	6.3.2.2.2
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 643, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 644, Gate Valve and Actuator, SIT Isolation	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 614, Gate Valve and Actuator, SIT Isolation	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 304, Gate Valve and Actuator, IRWST Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 305, Gate Valve and Actuator, IRWST Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 308, Gate Valve and Actuator, IRWST Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 309, Gate Valve and Actuator, IRWST Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 626, Globe Valve and Actuator, SI Isolation, CIV	6.3.2.2.3
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 646, Globe Valve and Actuator, SI Isolation, CIV	6.3.2.2.3

TABLE 3.11B-1 (Cont'd)

(Sheet 3 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 648, Globe Valve and Actuator, Check Valve Leakage Line Isolation	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 621, Globe Valve and Actuator, SIT Fill/Drain	6.3.2.2.2
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 623, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 624, Gate Valve and Actuator, SIT Isolation	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 331, Globe Valve and Actuator, Hot Leg Injection Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 321, Globe Valve and Actuator, Hot Leg Injection Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 628, Globe Valve and Actuator, Check Valve Leakage Line Isolation	6.3.2.2.5
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 633, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 634, Gate Valve and Actuator, SIT Isolation	6.3.2.2.2
Safety Injection	Varies	Varies	D, Subsphere	SI 602, Globe Valve and Actuator,Throttle, CIV	6.3.2.2.5
Safety Injection	Varies	Varies	D, Subsphere	SI 603, Globe Valve and Actuator, Throttle, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 636, Globe Valve and Actuator, SI Isolation, CIV	6.3.2.2.3
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 638, Globe Valve and Actuator, Check Valve Leakage Line Isolation	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 611, Globe Valve and Actuator, SIT Fill/Drain	6.3.2.2.2

TABLE 3.11B-1 (Cont'd)

(Sheet 4 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Safety Injection	Varies	Varies	A-1, A-2, Containment	SI 613, Globe Valve and Actuator, SIT Vent	6.3.2.2.2
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 616, Globe Valve and Actuator, SI Isolation, CIV	6.3.2.2.3
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 631, Globe Valve and Actuator, SIT Fill/Drain	6.3.2.2.2
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 390, Gate Valve and Actuator, Holdup Volume Spillway	6.8.2.2.3
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 391, Gate Valve and Actuator, Holdup Volume Spillway	6.8.2.2.3
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 392, Gate Valve and Actuator, Holdup Volume Spillway	6.8.2.2.3
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 393, Gate Valve and Actuator, Holdup Volume Spillway	6.8.2.2.3
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 394, Gate Valve and Actuator, Reactor Cavity Spillway	6.8.2.2.3
In-Containment Water Storage	Varies	Varies	A-1, A-2, Containment	SI 395, Gate Valve and Actuator, Reactor Cavity Spillway	6.8.2.2.3
Safety Injection	Short-Term	Short-Term	A-1, A-2, Containment	SI 618, Globe Valve and Actuator, Check Valve Leakage Line Isolation	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 302, Gate Valve and Actuator, SI Miniflow Return to IRWST, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 303, Gate Valve and Actuator, SI Miniflow Return to IRWST, CIV	6.3.2.2.5
Safety Injection	Varies	Varies	D, Subsphere	SI 609, Gate Valve and Actuator, Hot Leg Injection	6.3.2.2.5
Safety Injection	Varies	Varies	D, Subsphere	SI 604, Gate Valve and Actuator, Hot Leg Injection	6.3.2.2.5

TABLE 3.11B-1 (Cont'd)

(Sheet 5 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 300, Gate Valve and Actuator, CS/SCS IRWST Recirc Line Isolation, CIV	6.3.2.2.5
Safety Injection	Short-Term	Short-Term	D, Subsphere	SI 301, Gate Valve and Actuator, CS/SCS IRWST Recirc Line Isolation, CIV	6.3.2.2.5
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 600, Globe Valve and Actuator, SCS Train 2 Discharge Isolation, CIV	5.4.7.2.6
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 601, Globe Valve and Actuator, SCS Train 1 Discharge Isolation, CIV	5.4.7.2.6
Shutdown Cooling	Short-Term	Short-Term	A-1, A-2, Containment	SI 651, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.3
Shutdown Cooling	Short-Term	Short-Term	A-1, A-2, Containment	SI 652, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.3
Shutdown Cooling	Short-Term	Short-Term	A-1, A-2, Containment	SI 653, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.3
Shutdown Cooling	Short-Term	Short-Term	A-1, A-2, Containment	SI 654, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.3
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 655, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.6
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 656, Gate Valve and Actuator, SCS Suction Isolation, CIV	5.4.7.2.6
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 691, Globe Valve and Actuator, SCS Warm-Up Line Isolation	5.4.7.2.6
Shutdown Cooling	Short-Term	Short-Term	D, Subsphere	SI 690, Globe Valve and Actuator, SCS Warm-Up Line Isolation	5.4.7.2.6
Shutdown Cooling	Varies	Varies	D, Subsphere	SI 311, Globe Valve and Actuator, SDCHX Flow Control	5.4.7.2.6

TABLE 3.11B-1 (Cont'd)

(Sheet 6 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Shutdown Cooling	Varies	Varies	D, Subsphere	SI 310, Globe Valve and Actuator, SDCHX Flow Control	5.4.7.2.6
Shutdown Cooling	Varies	Varies	D, Subsphere	SI 313, Globe Valve and Actuator, SDCHX Bypass Flow Control	5.4.7.2.6
Shutdown Cooling	Varies	Varies	D, Subsphere	SI 312, Globe Valve and Actuator, SDCHX Bypass Flow Control	5.4.7.2.6
Containment Spray	Short-Term	Short-Term	D, Subsphere	SI 671, Gate Valve and Actuator, Containment Spray Header Isolation, CIV	6.5.2.2.3.1
Containment Spray	Short-Term	Short-Term	D, Subsphere	SI 672, Gate Valve and Actuator, Containment Spray Header Isolation, CIV	6.5.2.2.3.1
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 255, Globe Valve and Actuator, Seal Injection CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 505, Globe Valve and Actuator, RCP Bleed-off, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	A-1, A-2, Containment	CH 506, Globe Valve and Actuator, RCP Bleed-off, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	A-1, A-2, Containment	CH 515, Globe Valve and Actuator, Letdown Isolation	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	A-1, A-2, Containment	CH 516, Globe Valve and Actuator, Letdown Isolation	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	A-1, A-2, Containment	CH 575, Globe Valve and Actuator, Letdown Isolation, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	A-1, A-2, Containment	CH 560, Globe Valve and Actuator, RDT Suction Isolation, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 561, Globe Valve and Actuator, RDT Suction Isolation, CIV	9.3.4

TABLE 3.11B-1 (Cont'd)

(Sheet 7 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 580, Globe Valve and Actuator, RMW Supply to RDT Isolation, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 523, Globe Valve and Actuator, Letdown Isolation	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH 524, Globe Valve and Actuator, Charging Line Isolation, CIV	9.3.4
Chemical and Volume Control	Short-Term	Short-Term	D, Subsphere	CH-509, Gate Valve and Actuator, CVCS Makeup to IRWST, CIV	9.3.4
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 408, Gate Valve and Actuator	6.7.2.2.1
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 406, Globe/Angle Valve and Actuator	6.7.2.2.1
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 409, Gate Valve and Actuator	6.7.2.2.1
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 407, Globe/Angle Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 410, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 411, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 412, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 413, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 414, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 415, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 416, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 417, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 418, Globe Valve and Actuator	6.7.2.2.2
Safety Depressurization	Varies	Varies	A-1, A-2, Containment	RC 419, Globe Valve and Actuator	6.7.2.2.2
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 100, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 101, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 102, Gate Valve and Actuator	10.4.9.2.2.5

TABLE 3.11B-1 (Cont'd)

(Sheet 8 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 103, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 104, Globe Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 105, Globe Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 106, Globe Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 107, Globe Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 108, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 109, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 110, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 111, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 112, Gate Valve and Actuator	10.4.9.2.2.5
Emergency Feedwater	Varies	Varies	D, Subsphere	EF 113, Gate Valve and Actuator	10.4.9.2.2.5
Component Cooling Water	Continuous	Continuous	D, Nuclear Annex	CCWS Pump 1A and Motor	9.2.2.2.1.2
Component Cooling Water	Continuous	Continuous	D, Nuclear Annex	CCWS Pump 1B and Motor	9.2.2.2.1.2
Component Cooling Water	Continuous	Continuous	D, Nuclear Annex	CCWS Pump 2A and Motor	9.2.2.2.1.2
Component Cooling Water	Continuous	Continuous	D, Nuclear Annex	CCWS Pump 2B and Motor	9.2.2.2.1.2
Component Cooling Water	Short-Term	Short-Term	O, CCW Hx Structure	CC 100, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	O, CCW Hx Structure	CC 101, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annex	CC 102, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annex	CC 103, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 106, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 107, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 108, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 109, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 110, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 111, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 112, Throttle Valve and Actuator	9.2.2.2.1.9

TABLE 3.11B-1 (Cont'd)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Component Cooling Water	Varies	Varies	D, Subsphere	CC 113, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annex	CC 122, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 114, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 130, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 131, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 136, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 137, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 240, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 241, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 242, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 243, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	O, CCW Hx Structure	CC 200, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	O, CCW Hx Structure	CC 201, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annex	CC 202, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annex	CC 203, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 206, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 207, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 208, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	O, CCW Hx Structure	CC 209, Butterfly Valve and Actuator	9.2.2.2.1.9

TABLE 3.11B-1 (Cont'd)

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MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Component Cooling Water	Varies	Varies	D, Subsphere	CC 210, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 211, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Short-Term	Short-Term	D, Nuclear Annes	CC 222, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 212, Throttle Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 213, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 214, Butterfly Valve and Actuator	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 230, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 231, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	A-1, A-2, Containment	CC 236, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Component Cooling Water	Varies	Varies	D, Subsphere	CC 237, Butterfly Valve and Actuator, CIV	9.2.2.2.1.9
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Pump 1A and Motor	9.2.1.2.1.1
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Pump 1B and Motor	9.2.1.2.1.1
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Pump 2A and Motor	9.2.1.2.1.1
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Pump 2B and Motor	9.2.1.2.1.1
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Strainer 1A and Motor	9.2.1.2.1.5
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Strainer 1B and Motor	9.2.1.2.1.5
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Strainer 2A and Motor	9.2.1.2.1.5
Station Service Water	Continuous	Continuous	O, SSW Pump Structure	SSWS Strainer 1B and Motor	9.2.1.2.1.5
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 100, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 101, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 102, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 103, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 104, Plug Valve and Actuator	9.2.1.2.1.8

TABLE 3.11B-1 (Cont'd)

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MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND
WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 105, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 106, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 107, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 108, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 109, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 110, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 111, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 120, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 121, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 122, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 123, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 200, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 201, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 202, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 203, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 204, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 205, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 206, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 207, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 208, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 209, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 210, Plug Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 211, Plug Valve and Actuator	9.2.1.2.1.8

TABLE 3.11B-1 (Cont'd)

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MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 220, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 221, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 222, Butterfly Valve and Actuator	9.2.1.2.1.8
Station Service Water	Varies	Varies	O, SSW Pump Structure	SW 223, Butterfly Valve and Actuator	9.2.1.2.1.8
Essential Chilled Water	Continuous	Continuous	D, Nuclear Annex	ECW Refrigeration Unit	9.2.9.2.1
Essential Chilled Water	Continuous	Continuous	D, Nuclear Annex	ECW Refrigeration Unit	9.2.9.2.1
Essential Chilled Water	Continuous	Continuous	D, Nuclear Annex	ECW Circulation Pump and Motor	9.2.9.2.1
Essential Chilled Water	Continuous	Continuous	D, Nuclear Annex	ECW Circulation Pump and Motor	9.2.9.2.1
Diesel Gen. Fuel Oil	Continuous	Continuous	I, Diesel Building	Engine-Driven Fuel Oil Pump	9.5.4.2.1
Diesel Gen. Fuel Oil	Continuous	Continuous	I, Diesel Building	Engine-Driven Fuel Oil Pump	9.5.4.2.1
Diesel Gen. Fuel Oil	Continuous	Continuous	I, Diesel Building	Fuel Oil Booster Pump and Motor	9.5.4.2.1
Diesel Gen. Fuel Oil	Continuous	Continuous	I, Diesel Building	Fuel Oil Booster Pump and Motor	9.5.4.2.1
Diesel Gen. Cooling Water	Continuous	Continuous	I, Diesel Building	DGCW Circulation Pump	9.5.5.2.2
Diesel Gen. Cooling Water	Continuous	Continuous	I, Diesel Building	DGCW Circulation Pump	9.5.5.2.2
Diesel Gen. Cooling Water	Continuous	Continuous	I, Diesel Building	DGCW Keep Warm Pump and Motor	9.5.5.2.2
Diesel Gen. Cooling Water	Continuous	Continuous	I, Diesel Building	DGCW Keep Warm Pump and Motor	9.5.5.2.2
Diesel Gen. Cooling Water	Varies	Varies	I, Diesel Building	3-Way Thermostatic Control Valve	9.5.5.2.2
Diesel Gen. Starting Air	Varies	Varies	I, Diesel Building	Compressor and Motor	9.5.6.2.2
Diesel Gen. Starting Air	Continuous	Continuous	I, Diesel Building	Filter/Dryer Unit	9.5.6.2.2
Diesel Gen. Lube Oil	Continuous	Continuous	I, Diesel Building	Engine-Driven Lube Oil Pump	9.5.7.2.1
Diesel Gen. Lube Oil	Varies	Varies	I, Diesel Building	Prelube Oil Pump and Motor	9.5.7.2.2
Diesel Gen. Lube Oil	Varies	Varies	I, Diesel Building	Lube Oil Sump Tank Heater	9.5.7.2.2

TABLE 3.11B-1 (Cont'd)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND
WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Equip. & Floor Drainage	Varies	Varies	D, Subsphere	Reactor Bldg. Subsphere Sump Pump, A	9.3.3.2.2
Equip. & Floor Drainage	Varies	Varies	D, Subsphere	Reactor Bldg. Subsphere Sump Pump, B	9.3.3.2.2
Equip. & Floor Drainage	Varies	Varies	D, Subsphere	Reactor Bldg. Subsphere Sump Pump, C	9.3.3.2.2
Equip. & Floor Drainage	Varies	Varies	D, Subsphere	Reactor Bldg. Subsphere Sump Pump, D	9.3.3.2.2
Equip. & Floor Drainage	Varies	Varies	I, Diesel Building	Diesel Generator Bldg. Sump Pump	9.9.5.9.5.9
Equip. & Floor Drainage	Short-Term	Short-Term	A-1, A-2, Containment	Containment Sump Pump Discharge Line, CIV, Gate Valve and Actuator	9.3.3.2.1
Equip. & Floor Drainage	Short-Term	Short-Term	D, Subsphere	Containment Sump Pump Discharge Line CIV, Gate Valve and Actuator	9.3.3.2.1
Equip. & Floor Drainage	Short-Term	Short-Term	A-1, A-2, Containment	Containment Vent. Unit Condensate Drain Header CIV, Gate Valve and Actuator	9.3.3.2.1
Equip. & Floor Drainage	Short-Term	Short-Term	D, Subsphere	Containment Vent. Unit Condensate Drain Header CIV, Gate Valve and Actuator	9.3.3.2.1
Equip. & Floor Drainage	Short-Term	Short-Term	A-1, A-2, Containment	Reactor Drain Tank Gas Space To GWMS CIV, Globe Valve and Actuator	11.3
Equip. & Floor Drainage	Short-Term	Short-Term	D, Subsphere	Reactor Drain Tank Gas Space To GWMS CIV, Globe Valve and Actuator	11.3
Fuel Bldg. Ventilation	Continuous	Continuous	G, Fuel Building	Exhaust System Filter Train W/ Elec. Heater	9.4.2.2
Fuel Bldg. Ventilation	Continuous	Continuous	G, Fuel Building	Exhaust System Fan and Motor	9.4.2.2
Fuel Bldg. Ventilation	Varies	Varies	G, Fuel Building	Exhaust System Damper	9.4.2.2
Subsphere Bldg. Vent.	Continuous	Continuous	D, Nuclear Annex	Exhaust Fan and Motor	9.4.5

TABLE 3.11B-1 (Cont'd)

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MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Subsphere Bldg. Vent.	Continuous	Continuous	D, Nuclear Annex	Heating and Cooling Coil	9.4.5
Subsphere Bldg. Vent.	Continuous	Continuous	D, Nuclear Annex	Exhaust System Filter Train	9.4.5
Subsphere Bldg. Vent.	Varies	Varies	D, Subsphere	Ventilation Damper	9.4.5
Control Complex Vent.	Continuous	Continuous	J, Control Building	Air Handling Unit	9.4.1
Control Complex Vent.	Continuous	Continuous	J, Control Building	Battery Room Exhaust Fan	9.4.1
Control Complex Vent.	Varies	Varies	J, Control Building	Ventilation Damper	9.4.1
Diesel Bldg. Vent.	Continuous	Continuous	I, Diesel Building	Exhaust Fan and Motor	9.4.4
Diesel Bldg. Vent.	Varies	Varies	I, Diesel Building	Electric Heater	9.4.4
Diesel Bldg. Vent.	Continuous	Continuous	I, Diesel Building	Exhaust Fan and Motor	9.4.4
Diesel Bldg. Vent.	Varies	Varies	I, Diesel Building	Intake Ventilation Dampers	9.4.4
Diesel Bldg. Vent.	Varies	Varies	I, Diesel Building	Exhaust Ventilation Dampers	9.4.4
Annulus Vent.	Continuous	Continuous	D, Nuclear Annex	Filter Train W/ Elec. Heater	6.2.3.2
Annulus Vent.	Continuous	Continuous	D, Nuclear Annex	Exhaust Fan and Motor	6.2.3.2
Annulus Vent.	Varies	Varies	D, Nuclear Annex	Ventilation Damper	6.2.3.2
Station Serv. Wtr. Vent.	Continuous	Continuous	O, SSW Pump Structure	Supply Fan and Motor	9.4.8
Station Serv. Wtr. Vent.	Varies	Varies	O, SSW Pump Structure	Dampers	9.4.8
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 140, Main Steam Isolation Valve and Actuator	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 141, Main Steam Isolation Valve and Actuator	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 150, Main Steam Isolation Valve and Actuator	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 151, Main Steam Isolation Valve and Actuator	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 168, Main Steam Isolation Valve Bypass Valve and Actuator, CIV	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 169, Main Steam Isolation Valve Bypass Valve and Actuator, CIV	10.3.2.3.2.1

TABLE 3.11B-1 (Cont'd)

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MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 182, Main Steam Isolation Valve Bypass Valve and Actuator, CIV	10.3.2.3.2.1
Main Steam	Short-Term	Short-Term	M, Main Steam Valve House	SG 183, Main Steam Isolation Valve Bypass Valve and Actuator, CIV	10.3.2.3.2.1
Main Steam	Intermittent	Intermittent	M, Main Steam Valve House	SG 178, ADV and Actuator	10.3.2.3.2.3
Main Steam	Intermittent	Intermittent	M, Main Steam Valve House	SG 179, ADV and Actuator	10.3.2.3.2.3
Main Steam	Intermittent	Intermittent	M, Main Steam Valve House	SG 184, ADV and Actuator	10.3.2.3.2.3
Main Steam	Intermittent	Intermittent	M, Main Steam Valve House	SG 185, ADV and Actuator	10.3.2.3.2.3
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 132, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 174, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 137, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Steam	Varies	Varies	M, Main Steam Valve House	SG 105, ADV Isolation Valve and Actuator	10.3.2.3.2.3
Main Steam	Varies	Varies	M, Main Steam Valve House	SG 106, ADV Isolation Valve and Actuator	10.3.2.3.2.3
Main Steam	Varies	Varies	M, Main Steam Valve House	SG 107, ADV Isolation Valve and Actuator	10.3.2.3.2.3
Main Steam	Varies	Varies	M, Main Steam Valve House	SG 108, ADV Isolation Valve and Actuator	10.3.2.3.2.3
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 177, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 172, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7

TABLE 3.11B-1 (Cont'd)

(Sheet 16 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 130, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 175, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Main Feedwater	Short-Term	Short-Term	M, Main Steam Valve House	SG 135, Main Feedwater Isolation Valve and Actuator	10.4.7.2.7
Spent Fuel Pool Cooling	Continuous	Continuous	D, Subsphere	Pump and Motor	9.1.3.1.1
Combustible Gas Control	Continuous	N/A	D, Nuclear Annex	Hydrogen Recombiner	6.2.5.2.1
Combustible Gas Control	Continuous	N/A	D, Nuclear Annex	Hydrogen Analyzer	6.2.5.2.1.2
Combustible Gas Control	Short-Term	N/A	A-1, A-2, Containment	Div I Hydrogen Recombiner Suction From Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Combustible Gas Control	Short-Term	N/A	D, Subsphere	Div I Hydrogen Recombiner Suction From Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Combustible Gas Control	Short-Term	N/A	A-1, A-2, Containment	Div II Hydrogen Recombiner Suction From Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Combustible Gas Control	Short-Term	N/A	D, Subsphere	Div II Hydrogen Recombiner Suction From Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Combustible Gas Control	Short-Term	N/A	D, Subsphere	Div I Hydrogen Recombiner Discharge To Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Combustible Gas Control	Short-Term	N/A	D, Subsphere	Div II Hydrogen Recombiner Discharge To Containment, CIV, Globe Valve and Actuator	6.2.5.2.1
Breathing Air	Short-Term	Short-Term	D, Subsphere	Diaphragm Valve and Actuator, CIV	9.3.1.2.3
Station Air	Short-Term	Short-Term	D, Subsphere	Gate Valve and Actuator, CIV	9.3.1.2.2
Instrument Air	Short-Term	Short-Term	D, Subsphere	Diaphragm Valve and Actuator, CIV	9.3.1.2.1

TABLE 3.11B-1 (Cont'd)

(Sheet 17 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Containment Cooling & Vent.	Short-Term	Short-Term	A-1, A-2, Containment	High Volume Cont. Purge System Supply #1 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	High Volume Cont. Purge System Supply #1 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-1, A-2, Containment	High Volume Cont. Purge System Supply #2 CIV, Butterfly Valve Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	High Volume Cont. Purge System Supply #2 CIV, Butterfly Valve Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-1, A-2, Containment	High Volume Cont. Purge System Exhaust #1 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	High Volume Cont. Purge System Exhaust #1 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-1, A-2, Containment	High Volume Cont. Purge System Exhaust #2 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	High Volume Cont. Purge System Exhaust #2 CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	Low Volume Cont. Purge System Supply CIV, Butterfly Valve and Actuator	9.4.6
Containment Cooling & Vent.	Short-Term	Short-Term	A-1, A-2, Containment	Low Volume Cont. Purge System Supply CIV, Butterfly Valve and Actuator	9.4.6

TABLE 3.11B-1 (Cont'd)

(Sheet 18 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Containment Cooling & Vent.	Short-Term	Short-Term	A-3, Annulus	Low Volume Cont. Purge System Exhaust CIV, Butterfly Valve and Actuator	9.4.6
Process Sampling	Short-Term	Short-Term	A-1, A-2, Containment	SS 205, Pressurizer Steam Space Sampling Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	D, Subsphere	SS 202, Pressurizer Steam Space Sampling Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SS 203, Hot Leg 1 Sample Line CIV, Globe and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	SS 200, Hot Leg 1 Sample Line CIV, Globe and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	A-1, A-2, Containment	SS 210, Holdup Volume Line 1 Sample Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	A-1, A-2, Containment	SS 211, Holdup Volume Line 2 Sample Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	A-1, A-2, Containment	SS 204, Pressurizer Surge Line Sample Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	D, Subsphere	SS 201, Pressurizer Surge Line Sample Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Short-Term	Short-Term	D, Subsphere	SS 208, Holdup Volume Combined Sample Line CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-204, Steam Generator 1 Sample Line From Cold Leg CIV, Globe Valve and Actuator	9.3.2

TABLE 3.11B-1 (Cont'd)

(Sheet 19 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Process Sampling	Continuous	Continuous	D, Subsphere	SC-219, Steam Generator 1 Sample Line From Cold Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-211, Steam Generator 1 Sample Line From Hot Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	SC-228, Steam Generator 1 Sample Line From Hot Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-220, Steam Generator 1 Sample Line From Downcomer CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	SC-221, Steam Generator 1 Sample Line From Downcomer CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-222, Steam Generator 2 Sample Line From Cold Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	SC-223, Steam Generator 2 Sample Line From Cold Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-224, Steam Generator 2 Sample Line From Hot Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	SC-225, Steam Generator 2 Sample Line From Hot Leg CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	SC-226, Steam Generator 2 Sample Line From Downcomer CIV, Globe Valve and Actuator	9.3.2

TABLE 3.11B-1 (Cont'd)

(Sheet 20 of 21)

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Process Sampling	Continuous	Continuous	D, Subsphere	SC-227, Steam Generator 2 Sample Line From Downcomer CIV, Globe Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	Steam Generator 1 Combined Blowdown CIV, Gate Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	Steam Generator 1 Combined Blowdown CIV, Gate Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	A-1, A-2, Containment	Steam Generator 2 Combined Blowdown CIV, Gate Valve and Actuator	9.3.2
Process Sampling	Continuous	Continuous	D, Subsphere	Steam Generator 2 Combined Blowdown CIV, Gate Valve and Actuator	9.3.2
Fire Protection	Short-Term	Short-Term	D, Subsphere	Fire Protection Water Supply to Containment CIV, Gate Valve and Actuator	9.5.1.5.2
Fire Protection	Short-Term	Short-Term	D, Subsphere	Fire Protection Water Supply to Containment CIV, Gate Valve and Actuator	9.5.1.5.2
Normal Chilled Water	Short-Term	Short-Term	D, Subsphere	NCWS Supply To Cont. Vent. Units and CEDM Units CIV, Butterfly Valve and Actuator	9.2.9.2.2
Normal Chilled Water	Short-Term	Short-Term	D, Subsphere	NCWS Return From Cont. Vent. Units and CEDM Units CIV, Butterfly Valve and Actuator	9.2.9.2.2
Normal Chilled Water	Short-Term	Short-Term	A-1, A-2, Containment	NCWS Return From Cont. Vent. Units and CEDM Units CIV, Butterfly Valve and Actuator	9.2.9.2.2

MAJOR EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident[1] LOCA	MSLB	Specified Environmental[2] Conditions and Location	Equipment and Components[3]	Discussed in Section
Radiation Monitoring	Short-Term	Short-Term	A-1, A-2, Containment	Cont. Radiation Monitor (Inlet) CIV, Globe Valve and Actuator	11.5
Radiation Monitoring	Short-Term	Short-Term	D, Subsphere	Cont. Radiation Monitor (Inlet) CIV, Globe Valve and Actuator	11.5
Radiation Monitoring	Short-Term	Short-Term	A-1, A-2, Containment	Cont. Radiation Monitor (Outlet) CIV, Globe Valve and Actuator	11.5
Radiation Monitoring	Short-Term	Short-Term	D, Subsphere	Cont. Radiation Monitor (Outlet) CIV, Globe Valve and Actuator	11.5
Demineralized Water Makeup	Short-Term	Short-Term	D, Subsphere	CIV, Gate Valve and Actuator	9.2.3
Compressed Gas	Short-Term	Short-Term	D, Subsphere	Nitrogen Supply To SITs and RDT CIV, Globe Valve and Actuator	9.5.10
Reactor Coolant	Short-Term	Short-Term	A-1, A-2, Containment	RCP Oil Fill Line CIV, Gate Valve and Actuator	-
Reactor Coolant	Short-Term	Short-Term	D, Subsphere	RCP Oil Fill Line CIV, Gate Valve and Actuator	-

1. Definitions:

   Continuous - Component is required to operate throughout the design basis accident without interruption (i.e., up to six months).

   Short-term - Component is required to operate one time during the design basis accident (i.e., approximately a few seconds up to a few hours depending on the component and depending on the event).

   Intermittent - Component is capable of operating throughout the design basis accident (i.e., up to six months), starting and stopping on an as-needed basis.

   Varies - Component is capable of operating throughout the design basis accident (up to six months) depending on the situation, but it is not needed if something else can perform the same task.

2. Radiation environmental qualification requirements for individual components are developed as discussed in Section 3.11.6. Table 3.11A-1 provides the worst case upper bound radiation environment in the region where the component is located.

3. Electrical equipment listed in this Table will be qualified in accordance with the electrical equipment environmental qualification guidelines as stated in Section 3.11.2.

TABLE 3.11B-2

(Sheet 1 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

Component	Parameter	Module	Designation	Location	Environment (1) (2) Normal	Accident
Pressurizer	Pressure	Transmitter	PT-102 A, B, C, D	CB/SS	B	A-1, A-2
Pressurizer	Pressure	Transmitter	PT-103, 104, 105, 106	CB/SS	B	A-1, A-2
RCP	Pressure	Transmitter	PT-190A, 190B	CB/SS	B	A-1, A-2
Pressurizer	Level	Transmitter	LT-110A, 110B	CB/SS	B	A-1, A-2
RCS, T/C	Temperature	Element	TE-112, 122 CA, CB, CC, CD	CB/PS	B	A-1, A-2 NOTE (10)
RCS, T/H	Temperature	Element	TE-112, 122 HA, HB, HC, HD	CB/PS	B	A-1, A-2 NOTE (10)
RCP	Speed	Sensor & Cable	SE-113, 123, 133, 143 A, B, C, D	CB/PS	B	NOTE (8)
RCP	Speed	Transmitter	ST-113, 123, 133, 143 A, B, C, D	CB/SS	B	NOTE (8)
NI Safety	Power Channel	Detector	Chnl A, B, C, D	CB/MS	B	A-2
NI Safety	Power Channel	Preamp	Chnl A, B, C, D	RBS	C	D
CEDM	Position Indication	Reed Switch & Cable	CEDM 1-97	CB/MS	B	A-2
SI Tank	Pressure	Transmitter	PT-311, 321, 331, 341	CB/SS	B	A-1,A-2

TABLE 3.11B-2 (Cont'd)

(Sheet 2 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

Component	Parameter	Module	Designation	Location	Environment (1) (2) Normal	Accident
SI Tank	Level	Transmitter	LT-311, 321, 331, 341	CB/SS	B	A-1, A-2
SG 1	Level	Transmitter	LT-1113, 1114 A, B, C, D	CB/SS	B	A-1, A-2
SG 2	Level	Transmitter	LT-1123, 1124 A, B, C, D	CB/SS	B	A-1, A-2
SG 1	Pressure	Transmitter	PT-1013 A, B, C, D	CB/SS	B	A A-1, A-2
SG 2	Pressure	Transmitter	PT-1023 A, B, C, D	CB/SS	B	A-1, A-2
SG dP	Pressure Differential	Transmitter	PDT-115, 125 A, B, C, D	CB/SS	B	A-1, A-2
IRWST	Level	Transmitter	LT-350, 351	CB	B	A-1, A-2
IRWST	Temperature	Element	TE-350, 351	CB	B	A-1, A-2
EFWST 1	Level	Transmitter	Chnl A, C	NA	C	D
EFWST 2	Level	Transmitter	Chnl B, D	NA	C	D
EFW	Flow	Transmitter	Chnl A, B, C, D	RBS	C	D
EFW	Pressure	Transmitter	Chnl A, B, C, D	RBS	C	D
CCW	Flow	Transmitter	--	CCW Hx	N	O
CCW	Temperature	Element	--	CCW Hx	N	O
CS	Pressure	Transmitter	PT-338, 348	RBS	C	D
Containment	Pressure	Transmitter	PT-351, A, B, C, D	NA	C	D
Containment	Pressure	Transmitter	PT-352, A, B	NA	C	D
Containment	Pressure	Display	PR-351, 352	CR/MCB	J	J
Containment	Temperature	Element		CB	B	A-1, A-2
SIS	Flow	Transmitter	FT-311, 321, 331, 341	RBS	C	D
SDC	Pressure	Transmitter	PT-302, 305	RBS	C	D

TABLE 3.11B-2 (Cont'd)

(Sheet 3 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

Component	Parameter	Module	Designation	Location	Environment (1) (2) Normal	Accident
SDC	Flow	Transmitter	FT-306, 307	RBS	C	D
SDC	Temperature	Element	TE-300, 301	RBS	C	D
SDCHX	Temperature	Element	TE-305, 302	RBS	C	D
Containment Spray HX	Temperature	Element	TE-303x, 303y	RBS	C	D
Containment Spray Pump	Flow	Transmitter	FT-338, 348	RBS	C	D
Hot Leg Injection	Flow	Transmitter	FT-390, 391	RBS	C	D
Core Exit Thermocouple[5][9]	Temperature	Sensor	Chnl A, B	CB/MS, 19	B[6]	A-1, A-2
Heated Junction Thermocouple[5]	Temperature/ Level	Sensor	Chnl A, B	CB/MS	B[7]	A-1, A-2
Pressurizer	Level	Display	LR-110	CR/MCB	J	J
SDC	Temperature	Display	TR-300, 301	CR/MCB	J	J
RCS, T/H	Temperature	Display	TR-112HA 122HA	CR/MCB	J	J
RCP	Pressure	Display	PR-190A, B	CR/MCB	J	J
Steam Generator	Pressure	Display	PR-1013A, 1023A	CR/MCB	J	J
Steam Generator	Level	Display	LR-1113A, 1123A	CR/MCB	J	J
PPS Bistable	Trip	Processor	Chnl A, B, C, D	CR/PPSCC	J	J
PPS Interface	Test	Processor	Chnl A, B, C, D	CR/PPSCC	J	J
PPS Coincidence	Logic	Processor	Chnl A, B, C, D	CR/PPSCC	J	J
PPS Nuclear	Instrumentation	Panel	Chnl A, B, C, D	CR/APC	J	J
PPS ESFAS Initiation	Relay	Module	Chnl A, B, C, D	CR/PPSCC	J	J
PPS RPS Initiation	Relay	Module	Chnl A, B, C, D	CR/PPSCC	J	J
ESF CCS	Division	Cabinet	Chnl A, B, C, D	CR/ECCSC	J	J
ESF-CCS	Loop Controller	Enclosure	Chnl A, B, C, D	NA/RBS/ DGB	C/E/H	D/I

TABLE 3.11B-2 (Cont'd)

(Sheet 4 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

Component	Parameter	Module	Designation	Location	Environment (1) (2) Normal	Accident
RSPT	Output	Isolator	Chnl C, D	CR/CPCC	J	J
CPC	I/O	Module	Chnl A, B, C, D	CR/CPCC	J	J
CPC	CCU	Memory	Chnl A, B, C, D	CR/CPCC	J	J
CPC	Test	Panel	Chnl A, B, C, D	CR/CPCC	J	J
CEAC	Remote I/O	Multiplexor	Chnl A, B, C, D	RBS	C	D
CEAC	I/O	Module	Chnl B, C	CR/APC	J	J
CEAC	CPU	Memory	Chnl B, C	CR/APC	J	J
CPC/CEAC	Signal	Isolator	Chnl B, C	CR/APC	J	J
CEDM	Position Isolation	Assembly	Chnl A, D	CR/APC	J	J
RCPSSSS	Signal	Processor	Chnl A, B, C, D	CR/APC	J	J
Main Control	Panels	Assembly	Chnl A, B, C, D	CR	J	J
CPC	Operator's	Module	Chnl A, B, C, D	CR/MCB	J	J
DIAS-N	Display/ Alarm	Modules	N/A	CR/MCB	J	J
DIAS-N	Processing	Cabinets	N/A	CR	J	J
PPS	Operators	Module	Chnl A, B, C, D	CR/MCB	J	J
Reactor Trip System	Circuit	Breakers	Chnl A, B, C, D	NA	C	D
PPS Channel	Cabinet	Assembly	Chnl A, B, C, D	CR/PPSCC	J	J
APC	Cabinet	Assembly	Chnl A, B, C, D	CR	J	J
APS	Cabinet	Assembly	Chnl X, Y	CR	J	J
Remote Shutdown	Panel	Assembly	Chnl A, B, C, D	CR/RSP	J	J
NI Safety	Power Channel	Display	JR-001A	CR/MCB	J	J
Master Transfer	Switches	Module	Chnl A, B, C, D, X, Y	CR	J	J
DIAS-P	Display	Module	N/A	CR	J	J
PAMI	N/A	Processor	Chnl A, B	CR	J	J
HJTCS Heater Controller	N/A	Heater Controller Chassis	Chnl A, B		J	J

TABLE 3.11B-2 (Cont'd)

(Sheet 5 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

					Environment (1) (2)	
Component	Parameter	Module	Designation	Location	Normal	Accident
High Range Containment Area Monitor	Radiation	Monitor		CB	B	A-1, A-2
High Range Containment Area Monitor	Radiation	Monitor		CB	B	A-1, A-2
Main Steam Line Monitor	Radiation	Monitor		Main Steam Valve House	L	M
Main Steam Line Monitor	Radiation	Monitor		Main Steam Valve House	L	M
Unit Vent Monitor	Radiation	Monitor		NA	C	D
Unit Vent Post-Accident Monitor	Radiation	Monitor		NA	C	D
Control Room Air Intake Monitor	Radiation	Monitor		Control Building	J	J
Control Room Air Intake Monitor	Radiation	Monitor		Control Building	J	J
Containment Hydrogen Concentration	Hydrogen	Monitor		NA	C	D
Component Cooling Water HX Disch.	Flow	Transmitter	1A, 1B, 1C, 1D	CCW Hx/CR	N, J	O
SSW Pump, Outlet	Temperature	Element	1A, 1B, 2A, 2B	SSWPS	N	O

TABLE 3.11B-2 (Cont'd)

(Sheet 6 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

ABBREVIATIONS

Component

APS	Alternate Protection System
PCS	Power Control System
RCS	Reactor Coolant System
T/C	Reactor Inlet Pipe, T(cold)
T/H	Reactor Outlet Pipe, T(hot)
CEDM	Control Element Drive Mechanism
IRWST	In-containment Refueling Water Storage Tank
SDC	Shutdown Cooling
SDS	Safety Depressurization System
SDCHX	Shutdown Cooling Heat Exchanger
SG	Steam Generator
SI	Safety Injection
ESFAS	Engineered Safety Features Actuation System
CCWS	Component Cooling Water System
CPC	Core Protection Calculator
CS	Containment Spray System
CEAC	CEA Calculator
RCPSSSS	Reactor Coolant Pump Shaft Speed Sensing System
DPS	Data Processing System
PAMI	Post-Accident Monitoring Instrumentation
PPS	Plant Protection System
ESF-CCS	Engineered Safety Feature-Component Control System
NI	Nuclear Instrumention
APC	Auxiliary Process Cabinet
RSPT	Reed Switch Position Transmitter
EFW	Emergency Feedwater
EFWST	Emergency Feedwater Storage Tank
HJTCS	Heated Junction Thermocouples

TABLE 3.11B-2 (Cont'd)

(Sheet 7 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

APC	Auxiliary Process Cabinet	RSP
PAC	PCS Auxiliary Cabinet	PPSCC
CB	Containment Building	RBS
SSWPS	Station Service Water Pump Structure	MS
CR	Control Room or similar area with Class 1E Air Conditioning	PS
LO	Local	SS
MCB	Main Control Board	CCW Hx
NA	Nuclear Annex	ECCSC
DGB	Diesel Generator Building	
CPCC	Core Protection Calculator Cabinet	

TABLE 3.11B-2 (Cont'd)

(Sheet 8 of 8)

INSTRUMENTATION, IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS[11]

NOTES

(1) See Table 3.11A-1 for definition of environmental categories.

(2) Equipment located within a cabinet will be qualified allowing for temperature increase inside cabinet.

(3) Not used.

(4) Not used.

(5) Ex-vessel Portion of the instrument.

(6) Instrument Design life of 6 years.

(7) Instrument Design life of 10 years.

(8) Not qualified for accident environment.

(9) There is one core exit thermocouple for each ICI assembly.

(10) Only Channels A and B are qualified for accident environment.

(11) Radiation environmental qualification requirements for individual components are developed as discussed in Section 3.11.6. Table 3.11A-1 provides the worst case upper bound radiation environment in the region where the component is located.

TABLE 3.11B-3

(Sheet 1 of 5)

ELECTRICAL EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident: LOCA	MSLB	Specified Environmental Conditions & Location[1]	Equipment and Components	Remarks	Discussed in Section
Channel Related Vital Instrumentation Power & Control 120 VAC & 125 VDC	Continuous	Continuous	P, Battery Room	Enclosure & Components	Channels A, B, C, D	8.3.2.1.2.1
125 VDC Battery	Continuous	Continuous	P, Battery Room	Battery	One Per Channel (4)	8.3.2.1.2.1.2
120 VAC Inverter	Continuous	Continuous	P, Battery Room	Enclosure & Components	125 VDC - 120VAC Static Inv. One Per Channel (4)	8.3.2.1.2.1.3
125 VDC Dist. Center	Continuous	Continuous	P, Battery Room	Enclosure & Components	One Per Channel (4)	8.3.2.1.2.1.3
125 VDC Panelboard	Continuous	Continuous	P, Battery Room	Enclosure & Components	One Per Channel (4)	8.3.2.1.2.1.3
120 VAC Dist. Center	Continuous	Continuous	P, Battery Room	Enclosure & Components	One Per Channel (4)	8.3.2.1.2.1.4
125 VAC Panelboard	Continuous	Continuous	P, Battery Room	Enclosure & Components	One Per Channel (4)	8.3.2.1.2.1.4
Manual Transfer Switch	Continuous	Continuous	P, Battery Room	Enclosure & Components	Bus Tie To Other Charger	8.3.2.1.2.1.1
Auto Static Transfer Switch	Continuous	Continuous	P, Battery Room	Enclosure & Components	Bus Tie To Other Charger Within The Div., 125 VDC (4)	8.3.2.1.2.1.4
Manual Bypass Switch	Continuous	Continuous	P, Battery Room	Enclosure & Components	120 VAC Vital I & C Power One Per Channel (4)	8.3.2.1.2.1.4

TABLE 3.11B-3 (Cont'd)

(Sheet 2 of 5)

ELECTRICAL EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident: LOCA	Required Duration of Operation for Design Basis Accident: MSLB	Specified Environmental Conditions and Location[1]	Equipment & Components	Remarks	Discussed in Section
125 VDC Battery Charger	Continuous	Continuous	P, Battery Room	Enclosure & Components	125 VDC Vital Bat. Chargers One Per Channel (4)	8.3.2.1.2.1.1
125 VDC Battery Charger (Spare)	Continuous	Continuous	P, Battery Room	Enclosure & Components	125 VDC Vital Bat. Chargers One Fer Channel (4)	8.3.2.1.2.11
Division I & II Vital Instrumentation Power & Control 120 VAC & 125 VDC	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	Division I (Channels A & C) Division II (Channels B & D)	8.3.2.1.2.1
125 VDC Battery	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Battery	One Per Division (2)	8.3.2.1.2.1
120 VAC Inverter	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2) 125 VDC-120 VAC Static Inv.	8.3.2.1.2.1 and 8.3.2.1.2.1.3
125 VDC Charger	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.1
Manual Transfer Switch	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2) Bus Tie Between Div. I & II	8.3.2.1.2.1.1

TABLE 3.11B-3 (Cont'd)

(Sheet 3 of 5)

ELECTRICAL EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident: LOCA	MSLB	Specified Environmental Conditions and Location[1]	Equipment & Components	Remarks	Discussed in Section
Auto Static Transfer Switch	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2) Bus Tie Between Div. I & II	8.3.2.1.2.1.1
125 VDC Panelboard	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.3
125 VDC Dist. Center	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.3
120 VAC Panelboard	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.3
120 VAC Dist. Center	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.3
Manual Bypass Switch	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1.4
125 VDC Battery Charger (Spare)	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	One Per Division (2)	8.3.2.1.2.1 and 8.3.2.1.2.1.1
1E AC Power System 4160 VAC Aux. Power	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	Two Switchgear Assemblies Per Division (4)	8.3.1.1.2.1

TABLE 3.11B-3 (Cont'd)

(Sheet 4 of 5)

ELECTRICAL EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident: LOCA	MSLB	Specified Environmental Conditions and Location[1]	Equipment & Components	Remarks	Discussed in Section
4160/480 VAC Loadcenter	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	Two Loadcenters Per Division (4)	8.3.1.1.2.2
480 VAC Motor Control Centers	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	Four MCCs Per Division (8)	8.3.1.1.2.2
4160/480 VAC Transformer	Continuous	Continuous	K, 1E Elect. Equip. Rooms	Enclosure & Components	Two Loadcenters Per Division, (Standby)	8.3.1.1.2.2
Diesel Generator Control Panel	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	8.3.1.1.4
Diesel Engine Control Panel	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	8.3.1.1.4
Diesel Generator Neutral Grounding Cubicle	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	8.3.1.1.4
125 VDC Dist. Center	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	8.3.1.1.4
480 VAC Motor Control Center	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	8.3.1.1.4
Diesel Room Sump Pump Control Panel	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	9.5.9

TABLE 3.11B-3 (Cont'd)

(Sheet 5 of 5)

ELECTRICAL EQUIPMENT IDENTIFICATION, LOCATION AND WORST CASE ENVIRONMENTAL CONDITIONS

System	Required Duration of Operation for Design Basis Accident: LOCA	MSLB	Specified Environmental Conditions and Location[1]	Equipment & Components	Remarks	Discussed in Section
Fuel Oil Pump Booster Starter	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	9.5.4
Diesel Building HVAC Control Panels	Continuous	Continuous	I, Diesel Building	Enclosure & Components	One Per D/G (2)	9.4.4
Control Room HVAC Control Panels	Continuous	Continuous	J, Control Room	Enclosure & Components	One Per Train (2)	9.4.1.1
Fuel Building HVAC Control Panels	Continuous	Continuous	G, Fuel Building	Enclosure & Components	One Per Train (2)	9.4.2.1
480 VAC Motor Control Center	Continuous	Continuous	O, SSW Pump Structure	Enclosure & Components	One Per Train (2)	8.3.1.1.2.2
Distribution Panelboard	Continuous	Continuous	O, SSW Pump Structure	Enclosure & Components	One Per Train (2)	8.3.1.1.2.2
Station Service Water Pump Structure HVAC Control Panels	Continuous	Continuous	O, SSW Pump Structure	Enclosure & Components	One Per Train (2)	9.4.8.1
Subsphere Ventilation	Continuous	Continuous	D, Subsphere	Enclosure & Components	One Per Train (2)	9.4.3.1
Annulus Exhaust	Continuous	Continuous	A-1, A-2, Annulus	Enclosure & Components	One Per Train (2)	6.2.3
Battery Room HVAC Control Panels	Continuous	Continuous	J, Control Building	Enclosure & Components	Two Per Train (4)	9.4.1.1
Electrical Equipment Room HVAC Control Panels	Continuous	Continuous	J, Control Building	Enclosure & Components	Two Per Train (4)	9.4.1.1

Radiation environmental qualification requirements for individual components are developed as discussed in Section 3.11.6. Table 3.11A-1 provides the worst case upper bound radiation environment in the region where the component is located.

ATTACHMENT 3

ABB

ASEA BROWN BOVERI

Please Deliver To:

Name: Summer Sun

Company: NRC

Phone: Fax 301-504-2260

Message:

Attached are minor changes to Table 4.4-1 of CESSAR-DC per our discussion. None of these changes has any impact on safety. They are provided for your information to ensure FSER is consistent with CESSAR-DC. Any questions, please call.

Note: These changes will be included in Amendment V of CESSAR-DC.

From: Mark Kantrowitz

Phone: (203) 285-3255

Telefax: (203) 285-4441

Date: 2/22/94

Time:

This is Page 1, with 2 page(s) to follow.

ABB Combustion Engineering Nuclear Systems

Windsor, Connecticut 06095-500

TABLE 4.4-1
(Sheet 1 of 2)

THERMAL AND HYDRAULIC PARAMETERS

Reactor Parameters	System 80+	System 80	Waterford-3
Core Average Characteristics at Full Power:			
Total core heat output, MWt	3,914	3,800	3,390
Total core heat output, million Btu/h	13,360	12,970	11,570
Average fuel rod energy deposition fraction	0.974	0.974	0.974
Hot fuel rod energy deposition fraction	0.971	0.971	0.971
Primary system pressure, psia	2,250	2,250	2,250
Reactor inlet coolant temperature, °F	556	565	553
Reactor outlet coolant temperature, °F	615	~~624~~ 621	611
Core exit average coolant temperature, °F	617	623	613
Average core enthalpy rise, Btu/lbm	83	82	~~81~~ 80
Design minimum primary coolant flow rate, gpm	444,650	445,600	396,000
Design maximum core bypass flow, % of primary	3.0	3.0	~~3.5~~ 2.6
Design minimum core flow rate, gpm	431,300	432,200	385,700 ~~382,000~~
Hydraulic diameter of nominal subchannel, in.	0.471	0.471	0.471
Core flow area, ft^2	60.8	60.8	54.7
Core avg mass velocity, million $lbm/h\text{-}ft^2$	2.65	2.62	~~2.61~~ 2.64
Core avg coolant velocity, ft/s	16.7	16.8 ~~16.7~~	~~16.3~~ 16.5
Core avg fuel rod heat flux, $Btu/h\text{-}ft^2$	183,300	184,800[a)]	182,100
Total heat transfer area, ft^2	70,960	68,320[a)]	61,860

[a)]Corrected values for System 80 design

TABLE 4.4-1 (Cont'd)
(Sheet 2 of 2)
THERMAL AND HYDRAULIC PARAMETERS

Reactor Parameters	System 80+	System 80	Waterford-3
Average fuel rod linear heat rate kW/ft	~~5.37~~ 5.36	~~5.42~~ 5.41	~~5.34~~ 5.33
Power density, kW/liter	98.4	95.5	94.9
No. of active fuel rods	56,876	54,764	49,580
Power Distribution Factors:			
Rod radial power factor	1.55	1.55	1.55
Nuclear power factor	2.28	2.28	2.28
Total heat flux factor	~~2.35~~ 2.34	~~2.35~~ 2.34	~~2.35~~ 2.34
Engineering Factors:			
Engineering heat flux factor	1.03	1.03	1.03
Engineering enthalpy rise factor	1.03	1.03	1.03
Pitch, Bowing, and Clad Diameter Enthalpy Rise	1.05	1.05	1.05
Engineering factor on linear heat rate	1.03	1.03	1.03
Characteristics of Rod and Channel with Minimum DNBR:			
Maximum fuel rod heat flux, Btu/h-ft^2	~~432,200~~ 429,100	~~434,300~~[a] 432,700	~~427,900~~ 426,300
Maximum fuel rod linear heat rate, kW/ft	~~12.7~~ 12.6	12.7	12.5
UO_2 maximum steady state temperature, °F	3,179	3,205[a]	3,180
Outlet temperature, °F	644.1	645.7[a]	642
Outlet enthalpy, Btu/lbm	684.3	687.1[a]	680
Minimum DNBR at nominal conditions (CE-1 correlation)	2.00	1.98[a]	2.07

[a] Based on updated System 80 flow distribution

ATTACHMENT 4

L. Permitting no load to be carried over the loaded fuel racks whose impact energy, if dropped from the operating elevation, will exceed the impact energy of the postulated dropped fuel handling tool or the combination of the dropped fuel handling tool, fuel assembly, and any other handling component supported by the hoist cabling when lifting fuel assemblies. The Technical Specification incorporates the requirement that the impact energy of all loads carried over the loaded fuel racks will not exceed this condition.

Insert M

9.1.1.3.1.2 Criticality Safety Assumptions

The following assumptions are made in evaluating criticality safety:

A. Under postulated conditions of complete flooding by unborated water, the storage array is treated as a finite array of assemblies having an infinite fuel length.

B. Under postulated conditions of envelopment by aqueous foam or mist, a range of foam or mist densities is examined to ensure that the maximum reactivity of the array is established. The foam or mist is assumed to be pure water.

C. The poisoning effects of rack structure are neglected. Prior calculations have shown this to be a conservative assumption, where the degree of conservatism depends on the exact rack structure design. It is also assumed that no supplemental fixed poisons are utilized in the storage array.

D. A concrete storage cavity is utilized for new fuel storage. Two 11x11 rack modules are located in the cavity with cell blockers installed in alternate cells to limit new fuel storage to 121 fuel assemblies.

The criticality analyses for the new fuel racks assume a close-fitting, 2-foot thick concrete reflector on all six sides of the new fuel rack array. In actuality, the concrete walls surrounding the new fuel racks are separated from the racks by several inches, with the floor and material above the fuel also several inches away from the racks. A close fitting, thick concrete wall provides better neutron reflection than both the reflector consisting of a concrete wall separated from the array by several inches and the reflector consisting of the actual materials above the active fuel. Therefore, the configuration assumed for the criticality analyses is conservative with respect to the actual configuration of the new fuel rack array.

E. The rack is assumed to be filled to design capacity with fuel assemblies.

Insert M

M. Designing the refueling machine and spent fuel handling machine to hold their loads during a safe shutdown earthquake or a loss of power condition (See Section 9.1.4).

√

E. Meeting regulatory positions C.1 and C.2 of Regulatory Guide 1.29 and regulatory positions C.1 and C.6 of Regulatory Guide 1.13, as these positions relate to the ability of the components to withstand the effects of earthquakes.

Examples of compliance are demonstrated by the assignment of the various Seismic Categories to the building structures, fuel handling equipment, and other components as noted in Table 3.2-1 and the design of the equipment and components meeting these requirements. Fuel handling equipment that moves over the reactor core and spent fuel racks is also provided with seismic restraints to ensure that the components do not become disengaged from their operating rails and fall into the pool during a seismic event.

F. Meeting regulatory positions C.1, C.2 and C.3 of Regulatory Guide 1.13, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, and NUREG-0612 as they relate to radioactive release as a result of fuel damage.

Examples of compliance are demonstrated by the design of the fuel building which precludes movement of the spent fuel cask handling hoist over the new and spent fuel storage racks when they contain fuel assemblies, designation of load paths for all heavy lifts, limiting the weight and lift height of any load that is moved over the fuel racks such that its impact energy, if dropped, will not exceed the design impact energy of the fuel racks (See Section 9.1.2.3.1.1.G) or fuel pool, and ensuring that the lift height of the spent fuel shipping cask does not exceed 30 feet which limits the cask from being raised above the operating floor elevation.

G. Permitting no load to be carried over the loaded fuel racks whose impact energy, if dropped from the operating elevation, will exceed the impact energy of the postulated dropped fuel handling tool, fuel assembly, and any other handling component supported by the hoist cabling when lifting fuel assemblies. The Technical Specification incorporates the requirement that the impact energy of all loads carried over the loaded fuel racks will not exceed this condition.

H. Providing mechanical and electrical interlocks on the nuclear annex overhead hoists to preclude movement of fuel shipping containers or casks and other heavy loads from being transported over the spent fuel pool. (See Section 9.1.4.2.1.7)

Insert I

Y

9.1.2.3.1.2 Criticality Safety Assumptions

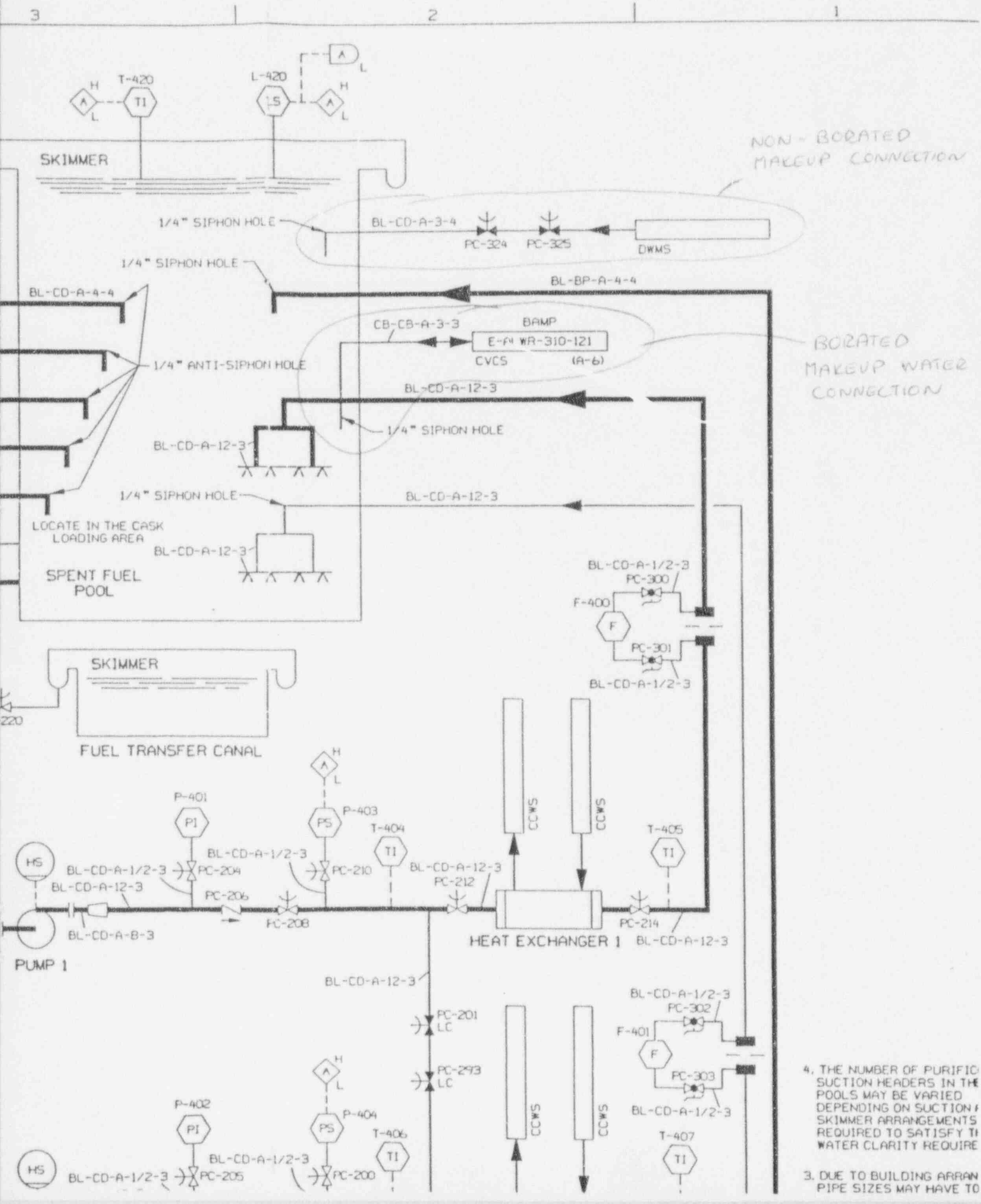
The following assumptions are made in evaluating criticality safety:

A. No control element assemblies (CEAs) are assumed to be present in the fuel assemblies.

Insert I

I. Designing the refueling machine and spent fuel handling machine to hold their loads during a safe shutdown earthquake or a loss of power condition (See Section 9.1.4). V

FIG 9.1-3 AMENDMENT 5
E-ALWR-310-140 REV 04



4. THE NUMBER OF PURIFIC
SUCTION HEADERS IN THE
POOLS MAY BE VARIED
DEPENDING ON SUCTION A
SKIMMER ARRANGEMENTS
REQUIRED TO SATISFY TH
WATER CLARITY REQUIRE

3. DUE TO BUILDING ARRAN
PIPE SIZES MAY HAVE TO

ATTACHMENT 5

9.4.1.2 System Description

The main control room air-conditioning system consists of two Divisions. Each Division has an outside air intake, louver, tornado damper, dampers, filtration unit, an air conditioning unit with fan, ducting, instrumentation and controls. Each redundant air conditioning unit consists of filter, safety-related chilled water coil for heat removal, electric heating coil and fan for air circulation. Each of the filtration units consists of prefilter, electric heater, absolute (HEPA) filter, carbon absorber, post filter (HEPA) and fan, along with ducts and valves and related instrumentation. Chilled water is supplied from the Essential Chilled Water System.

During normal operation, return air from the control room is mixed with a small quantity of outside air for ventilation, is filtered and conditioned in the control room air-conditioning unit, and is delivered to the control room through supply ductwork. Duct-mounted heating coils and humidification equipment provide final adjustments to the control room temperature and humidity for maintaining normal comfort conditions.

Each air inlet structure is provided with redundant radiation monitoring devices and a smoke detector. The designated MCR filtration units and ventilation fan start automatically on a Safety Injection Actuation Signal (SIAS) or high radiation signal. Upon failure of the designated filtration unit, the redundant filtration unit starts automatically. The MCR filtration unit filters particulates and potential radioactive iodines from ~~a portion~~ all of the return air, and delivers the filtered air to the inlet of the main air-conditioning unit.

The Technical Support Center air-conditioning system consists of an air-~~handling~~ conditioning unit, return air and smoke purge fans, and an emergency filter unit. The TSC is maintained at 1/8" water gauge positive pressure with respect to adjacent areas during post-accident conditions. A common supply air header and common outside air intake dampers are shared by the TSC and the control room to protect the TSC from the contaminants in the outside air intakes. The TSC can be isolated from the Main Control Room by using manual controls. The TSC is automatically isolated if control room pressurization falls below its design value.

The TSC is provided with shielding protection from direct radiation from an external radioactive cloud and internal radioactive sources. The combined effect of all radiation protection measures is designed to be adequate to limit the overall calculated radiation exposure to the personnel inside the TSC to the requirements of GDC 19. The computer room air-conditioning system consists of two 100% air-conditioning units and associated fans. Both the Technical Support Center and computer room air-handling systems are non-safety and non-seismic.

The balance of control complex air-conditioning systems consists of two redundant air-conditioning units, each with roughing filters, safety-related chilled water cooling coils and fans serving Division I electrical rooms, Channel A and Channel C. Two equal units are serving Division II Channel B and D. Each Division will function with one of the redundant air ~~handling~~ conditioning units delivering filtered, conditioned air to the various electrical equipment rooms including essential battery rooms. Chilled water is supplied from the Essential Chilled Water System. Each Division also contains redundant battery rooms with fan operating continuously to maintain the hydrogen concentration below two percent. Outlet ducts in battery rooms are located near ceiling for hydrogen control.

The Remote Shutdown Panel Room is located in the Division I area. Normally this room is cooled by the 70' Elevation Division I Electrical Equipment Room Air handling Unit. For redundancy purposes, the Remote Shutdown Panel Room is also cooled by a Division II- powered air handling unit which receives Division II Safety-related Chilled Water.

Return air from the various essential electrical equipment areas is mixed with a portion of outside air for ventilation, is filtered and conditioned in the air ~~handling~~ conditioning unit, and is delivered to the rooms through supply ductwork. Duct-mounted heating coils provide final adjustments to temperature in selected equipment rooms.

The Operation Support Center, personnel decon rooms, Break Room, Shift Assembly and Offices, Radiation Access Control and Cas. and Sec. Group areas all are served by an individual air conditioning unit consisting of a centrifugal fan, non-safety related chilled water coil and roughing filter. Two non-essential electrical and CEDM control rooms are served by two 100% air conditioning units consisting of a centrifugal fan, non-safety related chilled water coil and roughing filter. Each non-safety related electrical room A/C unit also serves non-safety related battery rooms and each of these battery rooms contains an exhaust fan operating continuously to maintain the hydrogen concentration below two percent.

As shown on Figure 9.4-2 all of these areas can receive outside air from the cleanest of two sources described for the control room. The roof exhaust fan shown serving the break room, personnel decon rooms, and shift assembly offices is actually located at least 80 feet from the outside air intake.

9.4.1.3 Safety Evaluation

The air-conditioning system serving the control room proper consists of two completely redundant, independent, full-capacity cooling systems. Each system is powered from independent, Class 1E power sources and headered on separate Essential Chilled Water Systems.

The safety-related and non-safety related battery rooms have hydrogen detection devices to monitor hydrogen concentration.

Indication of high radioactivity and toxic gas at outside air intakes is provided in control room.

Each Control Room Intake is provided with redundant, Seismic Category I, Class 1E, safety related radiation monitors. The CR air intake radiation monitors are located outside (upstream) of the Main CR intake dampers so that they can continue to monitor the air immediately outside the intakes to support the automatic selection capability. Upon detection of high radiation at either Control Room Intake or upon receipt of ~~the~~ Safety Injection Actuation Signal (SIAS), component control logic will automatically divert the control room air intake and recirculation air flows to pass through the designated Control Room Filtration Unit. Upon failure of the designated filtration unit to start, the redundant filtration unit will start automatically. At the same time, component control logic will isolate the Control Room Intake which has the greater radiation level and block the isolation of the Control Room Intake which has the lesser radiation level. These automatic features ensure that positive pressurization of the Control Room is maintained by uninterrupted pressurization air flow via the lesser contaminated Control Room Intake. Also, automatic selection logic is provided to continuously monitor and compare the radiation levels at both Control Room Intakes and effect Control Room Intake isolation damper realignments as needed so that the lesser contaminated Control Room Intake supplies pressurization air to the Control Room, even if radiation levels change. In addition, component control logic will ensure that the Control Room Intake isolation damper with the lesser radiation level is opened before the Control Room Intake isolation damper with the greater radiation level is closed. In the event of alignment failure, the operator is alerted by a Control Room alarm so that manual actions may be taken.

9.4.2 FUEL BUILDING VENTILATION SYSTEM

9.4.2.1 Design Basis

The Fuel Building Ventilation System is designed to:

A. Maintain a suitable environment for the operation, maintenance, and testing of equipment.

B. Maintain a suitable access and working environment for personnel.

All safety-related components of the mechanical equipment room cooling systems are designed as Seismic Category I equipment, and will remain functional following a design basis earthquake. Intake and exhaust structures are protected from wind-generated or tornado-generated missiles.

Redundant components of the safety-related mechanical equipment room cooling systems are physically separated and protected from internally generated missiles. When subjected to pipe break effects, the components are not required to operate because the served mechanical equipment is located in the same space as the cooling components. Therefore, a pipe break in the same mechanical safety train is the only possible means of affecting the cooling system.

The Subsphere Building essential HVAC exhaust filter trains are shown in Figure 9.4-5. The HEPA filters are designed to limit the offsite dose within the ~~guidelines~~ requirements of 10 CFR 100. The dose analysis for post accident releases from the subsphere only takes credit for the HEPA filters in the filter train. No credit is taken for the carbon adsorbers.

A differential pressure indicator controller located across the charcoal adsorber modulates a damper downstream of the filter train to maintain a constant system resistance as the filters load up. This arrangement assures a constant system flow. High and low differential pressure alarms provide indication of any abnormality in flow rates.

All safety-related components in the subsphere ventilation system are designed to permit in-service inspection.

Fresh air intakes are located in the control building duct shaft and are protected against adverse environmental conditions high winds, rain, snow, ice, etc.

The fresh air intakes for the Subsphere Building Ventilation System are located at least 30 feet above grade elevation to minimize intake of dust into the building. The fresh air intakes are provided with tornado dampers.

9.4.5.4 Inspection and Testing Requirements

Performance characteristics of the Subsphere Building Ventilation System will be verified through qualification testing of components as follows:

A. The safety-related equipment, fans, dampers, coils and ductwork will be designed and tested as outlined in Table 9.4-5.

B. One of each type of safety-related cooling fan will also be tested in accordance with AMCA.

TABLE 9.4-5

(Sheet 2 of 11)

DESIGN COMPARISON TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.52

Regulatory Guide 1.52 Position	System 80+
e. Components of systems connected to compartments that are unheated during a postulated accident should be designed for post-accident effects of both the lowest and highest predicted temperatures.	Complies.
2. System Design Criteria	
a. ESF atmosphere cleanup systems designed and installed for the purpose of mitigating accident doses should be redundant. The systems should consist of the following sequential components: (1) demisters, (2) prefilters (demisters may serve this function), (3) HEPA filters before the adsorbers, (4) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as metal zeolites), (5) HEPA filters after the adsorbers, (6) ducts and valves, (7) fans, and (8) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration.	Complies, except for Control Complex Ventilation System demisters are not provided. Water droplets will not be entrained in the airstream. Humidity control is provided by safety-related air-conditioning system which has provisions for both dehumidifying and heating to maintain relative humidity below 60%. Heaters are provided. in filtration unit.
b. The redundant ESF atmosphere cleanup systems should be physically separated so that damage to one system does not also cause damage to the second system. The generation of missiles from high-pressure equipment rupture, rotating machinery failure, or natural phenomena should be considered in the design for separation and protection.	Complies.
c. All components of an engineer-safety-feature atmosphere cleanup system should be designated as Seismic Category I (see Regulatory Guide 1.29) if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environments.	Complies.
d. If the ESF atmosphere cleanup system is subject to pressure surges resulting from the postulated accident, the system should be protected from such surges. Each component should be protected with such devices as pressure relief valves so that the overall system will perform its intended function during and after the passage of the pressure surge.	Not applicable. The systems are located outside of the containment and not exposed to pressure surges.

TABLE 9.4-1

(Sheet 1 of 18)

HVAC SYSTEM DESIGN PARAMETERS

Area or Location	Operational Mode: Normal	Operational Mode: Essential	Type System	Heat Load Btu/hr	Flow Rate/Unit: Air CFM	Flow Rate/Unit: Cool Water gpm	No Units % Capacity	Power Supply	Equipment
Control Room		X	Heat/Cool	300,000	6,000	70	2/100	460/120	Prefilter, cooling coil, fan, heat coil, humidifier
Control Room		X	Filter	-	~~2,000~~ 6,000	-	2/100	460/120	Filter train and fan
Control Room Mech Area		X	Cool	36,000	900	7	2/100	460/120	Prefilter, cooling coil and fan
Control Room	X		Smoke Fan	-	10,000	-	1/100	460/120	Fan
Tech Sup Mech	X		Filter	-	1,000	-	1/100	460/120	Filter train and fan
Tech Sup Mech	X		Smoke Fan	-	10,000	-	1/100	460/120	Fan

ATTACHMENT 6

14.3 CERTIFIED DESIGN MATERIAL

The Certified Design Material are those principal design characteristics, site parameters and interfaces, and the inspections, tests, analyses and acceptance criteria that are certified through the rulemaking process of 10 CFR Part 52 and are included in the formal Certification Rule. The selection criteria and processes used to develop the System 80+ Standard Plant Certified Design Material (CDM) are described in this section.

The System 80+ standard plant design information included in the CDM is derived from the more detailed design information presented in CESSAR-DC. The CDM is the most significant of the design information and reflects the tiered approach to design certification endorsed by the Commission [Staff Review Memorandum 2/15/91 regarding SECY-90-377; 10CFR Part 52 Statement of Considerations, 52 Federal Register 15372, 15377, (1989)]. In addition, the selection of the most significant design information was reviewed by multidiscipline design teams for completeness, accuracy, and consistency with the material in CESSAR-DC. Further, separate reviews were conducted by industry representatives and subsequently by combined industry/regulatory representatives in public session to ensure that the CDM met the criteria of "necessary and sufficient" as specified in 10 CFR 52.

The System 80+ standard plant Certified Design Material contains:

- An introduction section which defines terms used in the CDM, general provisions that are applicable to all CDM entries, and acronyms and legends used in the body of the CDM.

- Design descriptions for: a) systems that are fully within the scope of the System 80+ standard plant design certification, and b) the in-scope portion of those systems that are only partially within the scope of the System 80+ standard plant design certification. The intent of the CDM design descriptions is to delineate the principal design features and principal design characteristics that are referenced in the Design Certification Rule. The design descriptions are accompanied by the inspections, tests, analyses and acceptance criteria (ITAAC) required by 10 CFR 52.47(a) (1) (vi) to be part of the design certification application. The ITAAC define verification activities that are to be performed for a plant with specific pre-defined acceptance criteria to be met with the objective of confirming that the plant is built and will operate in accordance with the design certification. Successful completion of these ITAAC, together with the combined license (COL) applicant's ITAAC for the site-specific portions of the plant, will be the basis for NRC authorization to load fuel per the provisions of 10 CFR 52.103.

- Design descriptions and their associated ITAAC for design and construction activities that are applicable to more than one system. Design-related processes have been included in the CDM for:

  (1) Aspects of the System 80+ standard plant design likely to undergo rapid, beneficial technological developments in the lifetime of the design certification. Certifying the design processes associated with these areas of the design rather than specific design details permits applicants referencing the System 80+ standard plant design to take advantage of the improvements in technology available at the time of COL application and facility construction.

(2) Aspects of the design which are dependent upon characteristics of as-procured, as-installed systems, structures and components. These characteristics are not available at the time of certification and therefore cannot be used to develop and certify design details. However, the design processes associated with these aspects of the design can be certified and applied at the time of COL application and facility construction.

- Interface requirements as defined by 10 CFR Part 52.47(a) (1) (vii). Interface requirements are those requirements which must be met by the site-specific portions of the complete nuclear power plant that are not within the scope of the certified design. These requirements define characteristics of the site-specific features which must be provided in order for the certified design to comply with certification commitments. Interface requirements are defined for: a) systems entirely outside the scope of the design certification and b) the out-of-scope portions of those systems that are only partially within the scope of the design certification. The COL applicant will provide ITAAC for the site-specific design features that implement the interface requirements; therefore, the CDM does not include ITAAC for interface requirements.

- Site parameters used as the basis for the System 80+ standard plant design presented in CESSAR-DC. These parameters represent a bounding envelope of site conditions for any license application referencing the System 80+ standard plant design certification. ITAAC are not necessary for the site parameters entries because compliance with site parameters will be verified as part of issuance of a license for a plant that references the System 80+ standard plant design.

The following is a description of the criteria and methods by which specific technical entries for the CDM were selected. The structure of the description is based upon the structure of the CDM.

The criteria and methods that are discussed in the following sections are guidelines only. For some matters, the contents of the CDM may not directly correspond to these guidelines because special considerations related to the matters may have warranted an alternate, but essentially equivalent, approach. For such matters, a case-by-case determination was made regarding how or whether the matters should be addressed in the CDM. These determinations were based upon the principles inherent in Part 52 and its underlying purposes.

14.3.1 CDM SECTION 1.0: INTRODUCTION

Definitions, General Provisions, and Figure Legend and Abbreviations are described in this subsection.

14.3.1.1 CDM Section 1.1: Definitions

Selection Criteria - This Section defines terms which are used throughout the CDM and could (potentially) be subject to various interpretations. Selection of entries was based on the judgement that a particular word/phrase merited definition - with particular emphasis on terms associated with implementation of the ITAAC.

Selection Methodology - The terms defined in the Definition section were selected based on the perceived need to specifically state the context in which the term was to be used. These terms were identified during the preparation and review of the CDM.

Example Entries - Typical terms defined are "as-built," "Division," and "Type Test."

14.3.1.2 CDM Section 1.2: General Provisions

Selection Criteria - This section contains provisions that were selected on the basis that each provision was necessary to either a) define technical requirements applicable to multiple systems in the CDM or to b) provide clarification and guidance for implementation of the CDM.

Selection Methodology - Entries in the General Provisions section also were developed as part of the CDM definition and review process. Each entry is included to clearly state the general requirements, guidelines, and/or interpretations that are intended to be applied to the CDM.

Example Entries - Issues requiring general provisions treatment include guidance on interpretation of figures provided in the body of the CDM and defining the scope of what is included if a system configuration check is specified in an ITAAC entry.

14.3.1.3 CDM Section 1.3: Figure Legend and Abbreviation List

These were included only to aid a user of the CDM.

14.3.2 CDM SECTION 2.0: SYSTEM 80+ CERTIFIED DESIGN MATERIAL

This section of the CDM has the design description and ITAAC material for every system that is either fully or partially within the scope of the System 80+ standard plant design certification. The intent of this comprehensive listing of System 80+ standard plant systems is to define, at the CDM level, the full scope of the certified design.

Since preparation of system design descriptions and the associated ITAAC are sequential, separate processes, they are discussed separately in the next two subsections.

14.3.2.1 Design Descriptions

The Certified Design Description for each System 80+ standard plant system addresses the most significant design features and performance standards which pertain to the safety of the plant and include descriptive text and supporting figures. The intent of the CDM design descriptions is to define the System 80+ standard plant design characteristics which are referenced in the Design Certification Rule as a result of the certification provisions of 10 CFR Part 52.

Selection Criteria - The following criteria were considered in determining which information warranted inclusion in the certified design descriptions:

(1) The information in the certified design descriptions is to be derived only from the technical information presented in CESSAR-DC. This reflects the approach that the CDM contains the most significant design information and is based on the Commission directive in the Statement of Considerations for Part 52 (54 Fed. Reg. 15372, 15377 (1989)) that there "be less detail in a certification than in an application for certification." In this context, the "certification"

is the CDM and the "application for certification" consists of all the information in CESSAR-DC.

(2) The certified design descriptions contain only information from CESSAR-DC that is most significant to safety. CESSAR-DC contains a wide spectrum of information on various aspects of the System 80+ standard plant design, and not all of this information warrants inclusion in the certified design descriptions. This selection criterion reflects the Commission directive in the Statement of Considerations for Part 52 (Fed. Reg. 15372, 15377 (1989)) that the certified design should "encompass roughly the same design features that Section 50.59 prohibits changing without prior NRC approval." In determining what information is most significant to safety, several factors were considered, including the following:

(a) Whether the feature or function in question is necessary to satisfy the NRC's regulations in Parts 20, 50, 52, 73 and 100.

(b) Whether the feature or function in question pertains to a safety-related structure, system or component.

(c) Whether the feature or function in question is specified in the NRC's Standard Review Plan as being necessary to perform a safety-significant function.

(d) Whether the feature or function in question represents an important assumption or insight from the probabilistic risk assessment.

(e) Whether the feature or function in question is important in preventing or mitigating severe accidents.

(f) Whether the feature or function in question has had a significant impact on the safety or operation of existing nuclear power plants.

(g) Whether the feature or function in question is typically the subject of a provision in the Technical Specifications.

The absence or existence of any one of these factors was not conclusive in determining which information is significant to safety. Instead, these factors, together with the other factors listed in this section, were taken into account in making this determination.

(3) In general, only the safety-related features and functions of structures, systems and components are discussed in the certified design descriptions. Structures, systems, and components that are not classified as safety-related are discussed in the certified design descriptions only to the extent that they perform safety-significant functions or have features to prevent a significant adverse impact upon the safety-related functions of other structures, systems or components. This criterion follows from the principle that only features and functions that are safety-significant warrant treatment in the certified design. Non-safety-significant features and functions of safety-related structures, systems, and components are not generally discussed in the certified design descriptions.

(4) In general, the certified design descriptions for structures, systems, and components are limited to a statement of design features and functions. The design bases of structures, systems, and components, and explanations of their importance to safety, are provided in CESSAR-DC and are not included in the certified design descriptions. The purpose of the CDM design descriptions is to define the certified design. Justification that the design meets regulatory requirements is presented in CESSAR-DC. For example, the design descriptions for the emergency core cooling systems state the flow capacity of the systems; the descriptions do not provide information that demonstrates these flow capacities are sufficient to maintain post-accident fuel clad temperatures within 10 CFR 50 Appendix K limits.

(5) The certified design descriptions focus on the physical characteristics of the facility. The certified design descriptions do not contain programmatic requirements related to operating conditions or to operations, maintenance, or other programs because these matters are controlled by other means such as the Technical Specifications. For example, the design descriptions do not describe operator actions needed to control systems.

(6) The certified design descriptions in Section 2.0 of the CDM discuss the configuration and performance characteristics that the structures, systems, and components should have after construction is completed. In general, the certified design descriptions do not discuss the processes that will be used for designing and constructing a plant that references the System 80+ standard plant design certification. This is acceptable because the safety-performance of a structure, system, or component is demonstrated by appropriate inspections, tests and analyses on the as-built structures, systems and components. Exceptions to this criterion include:

(a) the welding, seismic qualification, environmental qualification and valve testing requirements addressed in CDM Section 1.2, and

(b) the various design and qualification processes defined in Section 3.

In addition, the programmatic aspects of the design and construction processes (training, quality assurance, qualification of welders, etc.) are part of the licensee's programs and are subject to commitments made at the time of COL issuance. Consequently, these issues are not addressed in the CDM.

(7) In general, the certified design descriptions address fixed design features expected to be in place for the lifetime of the facility. This is acceptable because portable equipment and replaceable items are controlled through operational-related programs. Since the CDM pertains to the design, it is not appropriate for it to include a discussion of these items. One exception to this general approach pertains to nuclear fuel, and control element assemblies (CEAs). These components are discussed in the certified design descriptions due to their importance to safety and the desire to control their overall design throughout the lifetime of a plant that references the System 80+ standard plant certified design.

(8) The certified System 80+ standard plant design descriptions do not discuss component types (e.g., valve and instrument types), component internals, or component manufacturers. This

approach is based on the premise that the safety function of a particular design element can be performed by a variety of component types and internals from different manufacturers. Consequently, a CDM entry that defines particular component type/manufacturer would have no safety-related benefits and would unnecessarily restrict the procurement options of future applicants and licensees. The CDM does contain exceptions to this general criterion, and these exceptions occur when the type of component is of safety-significance.

(9) The certified design descriptions do not contain any proprietary information.

(10) ~~In order to allow the applicant or licensee of a plant that references the System 80+ standard plant design certification to take advantage of improvements in technology, the certified design descriptions in general do not prescribe design features that are the subject of rapidly evolving technology. Examples are: specific hardware configuration of the main control room and instrumentation and control systems. This issue is discussed further in Section 14.3.3.~~

(10)
~~(11)~~ The CDM design description is intended to be self-contained and does not make direct reference to CESSAR-DC, industrial standards, regulatory requirements or other documents. (There are some exceptions involving the ASME Code and the Code of Federal Regulations.) If these sources contain technical information of sufficient safety-significance to warrant CDM treatment, the information has been extracted from the source and included directly in the appropriate system design description. This approach is appropriate because it is unambiguous and it avoids potential confusion regarding how much of a referenced document is encompassed in, and becomes part of, the CDM.

Insert B

Selection Methodology - Using the criteria listed above, design description material was developed for each system by reviewing the CESSAR-DC material relating to that system.

Of particular importance was the review of those sections of CESSAR-DC that document plant safety evaluations showing acceptable plant performance. Specifically, detailed reviews were conducted of the following in chapters of CESSAR-DC; the flooding analyses in Chapter 5, the analysis of overpressure protection in Chapter 5, containment analyses in Chapter 6, the core cooling analyses in Chapters 6 and 15, the analysis of fire protection in Chapter 9, the safety analysis of transients and anticipated transients without scram (ATWS) in Chapter 15, the radiological analyses in Chapter 15, and the resolution of unresolved or generic safety issues and Three Mile Island issues in Chapters 1 and Appendix A. These reviews were a key factor in identifying the important, safety-related system design information warranting discussion in the design descriptions.

Example Entries - Because the safety significance of the System 80+ standard plant systems varies considerably, application of the criteria listed above results in a graded treatment of the systems. This leads to considerable variations in the scope of the design description entries. The following lists the types of System 80+ standard plant systems and is a summary of the overall consequences of this graded treatment:

~~Insert A (Cont'd)~~

INSERT B

~~CESSAR INPUT TO SECTION 14.3~~

~~Section 14.3.3 CDM Section 3.0: Additional Certified Design Material~~

~~14.3.3.3 Initial Test Program~~

The ITP is addressed in Section 2.0 of the CDM and is

The CE Initial Test Program (ITP) defines testing activities that will be conducted following completion of construction and construction related inspections and tests. The ITP extends through to the start of commercial operation of the facility. ~~This~~ The program ~~is extensively discussed~~ detailed in Chapter 14 of the CESSAR ~~and centers heavily on testing of the CE System 80+ safety related systems.~~ The testing specified in Section 2.0 and Section 3.0 ~~of the~~ CDM are a subset of the ITP.

~~A summary of the ITP has been included in CDM Section 2.11. This summary includes an overview of the ITP structure together with commitments related to test documentation and administrative controls.~~ ~~This information~~ The ITP has been included in the CDM because of the importance of the ITP in defining comprehensive ~~pre and post-fuel load~~ testing for the as-built facility to demonstrate compliance with the design certification. ~~Key pre fuel load ITP testing for individual systems is defined in the system ITAAC in CDM Sections 2 and 3.~~ in accordance with detailed procedures and administrative controls.

No ITAAC entries ~~have been included~~ are necessary in the CDM for the ITP. This is acceptable because:

(a) ~~Many of the~~ These ITP activities that involve testing with the reactor containing fuel or conducted at various power levels ~~and thus~~ cannot be completed prior to fuel load. (Part 52 requires ITAAC to be completed prior fuel load).

(b) Testing activities specified as part of the ITAAC ~~to~~ in the CDM ~~Section 2~~ must be performed prior to fuel load. Since ~~these~~ the ITAAC ~~testing activities~~ address the design features and characteristics of key safety significance, additional ITAAC ~~for~~ related to the ITP ~~as defined in Section 2.11~~ are not necessary to assure that the as-built plant conforms with the certified design.

FSER INPUT TO FSER SECTION 14.3

See attached FSER markups, similar to those submitted to you for the ABWR FSER.

System Type	Scope of Certified Design Description
Safety-related systems that contribute to plant performance during design basis accidents (e.g., emergency core cooling systems).	Major safety-related features and performance characteristics.
Non-safety-related systems involved in beyond-design-basis events (e.g., combustion turbine generator contribution to station blackout event sequence).	Brief discussion of design features and performance characteristics affecting the safety of the plant's response to the event(s).
Non-safety-related systems potentially impacting safety (e.g., potential missiles from the main turbine).	Brief discussions of design features which prevent or mitigate the potential safety concern.
Non-safety-related systems which affect overall plant design (e.g., Chemical and Volume Control System)	Case-by-case evaluation. A brief discussion of the system if warranted by overall standardization goals.
Non-safety-related systems with no relationship to safety or any influence on overall plant design (e.g., Turbine Building Service Water System).	Limited description of system features.

For safety-related systems, application of the above criteria resulted in design description entries which include the following information, as applicable: The name and scope of the system; purpose; safety-related modes of operation; system's classification (i.e., safety-related, seismic category, and ASME Code Class); location; the basic configuration of safety-significant components (usually shown by means of a figure); the type of electrical power provided; the electrical independence and physical separation of divisions within the system; important instruments, controls, and alarms located in the Main Control Room; identification of Class 1E electrical equipment qualified for its intended environment; motor-operated valves that have an active safety-related function; and other functions that are significant to safety.

The certified design descriptions for non-safety-related systems also include the information listed above, but only to the extent that the information is relevant to the system and has a significance to safety. Since much of this information is not relevant to non-safety-related systems, the certified design descriptions for non-safety-related systems are generally substantially less extensive than the descriptions for safety-related systems.

14.3.2.2 Inspections, Tests, Analyses and Acceptance Criteria

A table of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) entries is generally provided for each system containing design description entries. The intent of these ITAAC is to define activities that will be undertaken to verify the as-built system conforms with the design features and characteristics defined in the corresponding CDM design description for that system. At the time of fuel loading, following certification of completion of the ITAAC by the referencing COL holder, the ITAAC become archival in that they have no follow-on regulatory status for that referencing COL holder.

A three-column table format is used to specify the [1] design commitment, [2] inspections, tests, and analyses, and [3] acceptance criteria for each ITAAC. Each design commitment in the left-hand column of the ITAAC has one or more associated inspection, test or analysis (ITA) requirement that is specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

Selection Criteria - The following were considered when determining which information warranted inclusion in the CDM ITAAC entries:

- The scope and content of the ITAAC correspond to the scope and content of the certified design descriptions. There are no ITAAC for those aspects of the design that are not addressed in the design description. This is appropriate because the objective of the ITAAC design certification entries is to verify that the as-built facility has the design features and performance characteristics defined in the design descriptions.

- With only a few special-case exceptions (e.g., initial test program), each System 80+ standard plant system with a design description text has an ITAAC table with one or more entries. This reflects the assessment that, in general, design features meriting a CDM description also merit an ITAAC entry to verify that the feature has been included in the as-built facility.

- One inspection, test, or analysis may verify one or more provisions in the certified design description. In particular, an ITAAC which calls for a system functional test or an inspection of basic configuration may verify a number of provisions in a certified design description. Therefore, there is not necessarily a one-to-one correspondence between the ITAAC and the certified design descriptions. In certain circumstances, documentation that verifies compliance of an inspection, test or analysis at one plant may be used as a basis to demonstrate compliance at one or all subsequent plants without repeating that inspection, test or analysis. For example, type testing of valves.

- As required by 10 CFR 52.103, the inspections, tests, and analyses must be completed (and the acceptance criteria satisfied) prior to fuel loading. Therefore, the ITAAC do not include inspections, tests, or analyses that are dependent upon conditions that only exist after fuel load.

- In general, the ITAAC verify the as-built configuration and performance characteristics of structures, systems and components as identified in the CDM design descriptions. With limited exceptions (e.g., welding), the ITAAC do not address typical construction processes for the reasons discussed in item (6) of Section 14.3.2.1. As necessary, ITAAC coverage of the exceptions is by:

  (1) The provisions of CDM Section 1.2, Items (1) through (4) that are invoked by configuration verification entries in individual system ITAAC tables.

  (2) The ITAAC entries in Section 3 of the CDM.

identified in the CDM design descriptions. Also, in some cases, CESSAR-DC has identified detailed criteria applicable to the same design feature or function that is the subject of more general acceptance criteria in the ITAAC table.

Ranges, limits, and/or tolerances are included for numerical AC. This is necessary and acceptable because:

- Specification of a single-value AC is impractical since minute/trivial deviations would represent noncompliance.
- Tolerances recognize that as-built variations can occur which do not affect function or performance.
- Minor variations within the tolerance bounds have no impact on plant safety.

14.3.3 CDM SECTION 3.0: ADDITIONAL CERTIFIED DESIGN MATERIAL

Entries in this section of the CDM have the same structure as the system material discussed in Section 14.3.2; i.e., design description text and figures and a table of ITAAC entries. The objective of this CDM material is to address selected design and construction activities which are applicable to more than one system and cannot appropriately be covered in the system-by-system information presented in Section 2.0 of the CDM. There are only two ~~entries in Section~~ Piping Design and Radiation Protection 3.0 of the CDM, ~~and the following summarizes the scope and bases for these entries~~. INSERT A ~~For each, the design description text defines the applicability of the entry.~~

The following summarizes the scope and bases for the Piping Design and Radiation Protection entr

14.3.3.1 Radiation Protection

The radiation protection section of the CDM defines the processes by which it will be confirmed that the as-built facility has radiation protection features that maintain exposures for both plant personnel and the general public below allowable limits. The material applies to the radiological shielding and ventilation design of buildings within the scope of the certified design.

Certification of plant radiation protection features via process definition rather than via certification of specific design features is necessitated and justified by the following:

(1) Actual radiological source terms are dependent upon the characteristics of the as-built, as-installed equipment. For example, such parameters as equipment sizes, geometry, and valve stem leakage rates influence source terms. Consequently, final radiological evaluation cannot be completed prior to availability of this as-built data and therefore cannot be used to finalize radiological protection design features at the time of design certification.

(2) Radiological studies using representative design assumptions have been completed and reported in the CESSAR-DC Chapter 12. These preliminary studies show the radiological protection features are such that acceptance criteria related to occupational and general public exposure are met. This provides high confidence that the processes defined in the radiological CDM entry can be successfully executed within the envelope of the certified design. This confidence

Selection Methodology - Using the criteria listed above, ITAAC table entries were developed for each system. This was achieved by evaluating the design features and performance characteristics defined in the CDM design description and preparing an ITAAC table entry for each design description entry that satisfied the above selection criteria. As a result of this process, there is a close correlation (although not necessarily one-for-one for the reasons noted in item (2) above) between the left-hand column of the ITAAC table and the corresponding design description entry.

Having established the design features for which ITAAC are appropriate, the ITAAC table was completed by selecting the method to be used for verification (either a test, an inspection or an analysis (ITA) or a combination of inspection, test, and analysis) and the acceptance criteria (AC) against which the as-built feature or functional performance will be measured. Selection of these items is dependent upon the plant feature to be verified but was guided by the following:

Inspection — To be used when verification can be accomplished by visual observations, physical examinations, review of records based on visual observations or physical examinations that compare the as-built structure, system or component condition to one or more design description commitments.

Test — To be used when verification can be accomplished in a practical manner by the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of the as-built structures, systems or components. The type of tests identified in the ITAAC tables are not limited to in-situ testing of the completed facility but also include (as appropriate) other activities such as factory testing, special test facility programs, and laboratory testing.

Analysis — To be used when verification can be accomplished by calculation, mathematical computation or engineering or technical evaluations of the as-built structures, systems or components. (In this case, engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems or components.)

The proposed verification activity is identified in the middle column of the ITAAC table. In some cases, CESSAR-DC also provides details regarding implementation of the verification activity. For example, CESSAR-DC Chapter 14 test abstracts contain specific testing descriptions related to ITAAC. This CESSAR-DC information is not referenced in the CDM and is not part of the CDM; it is considered as providing only one of potentially several acceptable methods for completing the ITA.

Selection of acceptance criteria (AC) is dependent upon the specific design characteristic being verified by the ITAAC table entry; in most cases, the appropriate AC is based upon the CDM design description. For many of the ITAAC, the AC is a statement that the as-built facility has the design feature or performance characteristic identified in the design description. A central guiding principle for AC preparation is the recognition that the criteria should be objective and unambiguous. The use of objective and unambiguous terms for the AC will minimize opportunities for multiple, subjective (and potentially conflicting) interpretations as to whether an AC has, or has not, been met. In some cases, the ITAAC acceptance criteria contain parameters from CESSAR-DC that are not specifically

INSERT A (Section 14.3.3)

Instrumentation and Controls (I&C), and Human Factor Engineering (HFE) are not specified in this section, but are included as defined systems in Section 2.0 of the CDM.

The Instrumentation and Controls design is specified as defined systems, e.g., Plant Protection System, in Section 2.0 of the CDM in the same manner as other system, structures, and components. The I&C system designs including applicable program plans, e.g. ~~the~~ Software Development Plan, have been completely and approved. The aspects of the design which are not completely specified relate to the ~~hardware~~ components to be used in the as-built system. For these aspects, detailed plans have been developed, reviewed, and approved. This includes a software program manual governing verification and validation activities, an equipment qualification plan, and a plan specifying safety system dedication of commercial products. This level of design detail combined with the completion of required detailed planning documents provides the bases for a positive safety determination and the ability to specify ITAAC to assure that the as-built I&C system conform to the certified design. Improvements in I&C technology are still readily accommodated into the I&C systems at the component level without affecting the certified design.

Insert A (Continued)

Human factors design is incorporated in the design of the Main Control Room (MCR) and the Remote Shutdown Room (RSR) which are contained in Section 2.0 of the CDM. Design details, features, and characteristics including applicable planning documents are completed such that only human factors verification and validation of the as-built configurations of the MCR and RSR are required to complete specified ITAAC. Design details for the MCR configuration, integrating display (IPSO), and six standard man-machine interface features used throughout the MCR and RSR designs were reviewed and approved. The human factors process review included the eight HFE Program Review Model elements. Four PRM elements were completed. Procedures development, the fifth element, is performed by a COL applicant. The remaining three elements were addressed with detailed plans and human factors guidance documents which were reviewed and approved. Consequently, the ITAAC specified in the CDM for the MCR and the RSR relate only to the human factors verification and validation evaluations of the as-built configurations with the detailed evaluation methods and acceptance criteria specified in CESSAR DC and its referenced plans. This level of detail in the design and the completeness of the supporting plans and guidance documents provide the bases for a positive safety determination based on the specified ITAAC.

is based in part on the recognition that technology associated with radiation sources and protection is well understood and design methodology and protection technology would only improve during the lifetime of the design certification.

Selection of entries in the CDM utilized the same selection criteria and methodology as discussed above for the Section 2.0 system entries.

14.3.3.2 Piping Design

The piping design section of the CDM defines the processes by which System 80+ standard plant piping will be designed and evaluated. The material applies to piping systems that are classified as nuclear safety-related. In general, these piping systems are designated as Seismic Category I and are further classified as ASME Code Section III, Class 1, 2 or 3. The section also addresses the consequential effects of pipe rupture such as jet impingement, potential missile generation, and pressure/temperature effects.

Certification of plant safety-related piping systems via design processes rather than via certification of specific design features is necessitated and justified by the following:

(1) Piping design is based on detailed piping arrangement information as well as the geometry and dynamic characteristics of the as-procured equipment that forms part of the piping system. This detailed plant-specific information is unavailable at the time of design certification and cannot therefore be used to develop detailed design information. This precludes certification of specific piping designs.

(2) An extensive definition of design methodologies is contained in Chapter 3 of CESSAR-DC. These methodologies are not considered to be part of the CDM but are one of several methods for executing the design process steps defined in the piping design CDM. In addition, sample design calculations have been performed with these methods to provide confidence that they are complete and yield acceptable design information.

(3) Piping design for nuclear plants is a well-understood process based on straightforward engineering principles. This, together with the methodology definition and sample calculations, provides confidence that future design work by individual applicants/licensees will result in acceptable designs that properly implement the applicable requirements.

The technical material in the piping design CDM entry was selected using the criteria and methodology as discussed above for the Section 2.0 system entries.

14.3.4 CDM SECTION 4.0: INTERFACE REQUIREMENTS

This section of the CDM provides interface requirements for those structures, systems and components of a complete power-generating facility that are either totally or partially not within the scope of the System 80+ standard plant design as defined in the certification application. For the System 80+ standard plant, these systems are identified in Section 1.9. Generally, structures, systems and components that are part of, or within, the Nuclear Island Structure, Turbine Building, Radwaste Building, Diesel Fuel Storage Building, and Component Cooling Water Heat Exchanger Structure are in the System 80+ standard plant scope. Those portions of the plant outside of these buildings are not generally in the System 80+ standard plant scope. This scope split occurs because design of the plant features located outside the main buildings is dependent upon site-specific characteristics which are not specified at the time of certification (e.g., the source of plant cooling water, the characteristics of the electrical grid to which the plant is connected, etc.). The basis for this interface requirements entry in the CDM is the discussion in 10 CFR 52.47(a) (1) (vii). An applicant for a license that references the CESSAR-DC design certification must provide site-specific systems with design features/characteristics that comply with the interface requirements.

An entry is provided in Section 4.0 of the CDM for each of the systems listed in CESSAR-DC Section 1.9; for systems that have no interface requirements of sufficient safety-significance to warrant CDM treatment, there are no entries. For systems that are partially within the scope of the System 80+ standard plant, interface requirements are listed in CDM Section 4.0 and in a separate sub-part of the Section 2.0 entry which addresses the in-scope portion of the system. In all cases, the CDM entries for these systems are limited to defining interface requirements. Conceptual designs for the out-of-scope interfacing systems are required by 10 CFR Part 52.47(a) (1) (ix); these designs are presented in CESSAR-DC but are not addressed in the CDM. This is appropriate because the applicant will provide site-specific designs that meet the interface requirements; these site-specific designs may not, and need not, correspond to the conceptual designs described in CESSAR-DC. The CDM does not define any ITAAC associated with the interface requirements. This is acceptable because ITAAC for the plant structures, systems, and components outside the scope of the System 80+ standard plant design certification will be provided on a site-specific, design-specific basis by the individual COL applicants who reference the System 80+ standard plant design certification. (Part of the review process at the time of the license application will be to assess compliance of the site-specific designs with the interface requirements.)

10 CFR Part 52.47(a) (1) (viii) specifies that design certification applications contain justification that the requirements are verifiable through inspection, testing or analysis and that the method to be used for verification be included as part of the ITAAC. The introductory text of CDM Section 4.0 addresses these issues by stating the interface requirements are similar in nature to the design commitments in Section 2.0 for which ITAAC have been developed. This represents justification that a COL applicant will be able to develop ITAAC to verify compliance with the design features or characteristics that implement the interface requirements. The methods to be used for these verifications will be specified in the COL ITAAC and will be similar to the methods in the Section 2.0 ITAAC for comparable/similar design characteristics.

Selection Criteria - The selection criteria listed in Section 14.3.2.1 were used to guide selection of interface requirements defined in Section 4.0 of the CDM (or in the Section 2.0 entries referenced from Section 4.0). The intent is that the interface requirements in the CDM define key, safety-significant design attributes and performance characteristics of the site-specific, out-of-scope portion of the plant which must be provided in order for the certified portions of the System 80+ standard plant to comply with the design commitments in the CDM. It is an objective of this section that it address interfaces between in-scope and out-of-scope portions of the plant that are unique to the System 80+ standard plant design; it is not intended that it be a comprehensive listing of design requirements applicable to the out-of-scope portions of the plant. A discussion of the design feature of out-of-scope portions of the plant will be provided for NRC review when the COL applicant submits a site-specific safety analysis report.

Selection Methodology - The interface requirements included in the CDM were selected from the interface requirements listed in the CESSAR-DC for fully or partially out-of-scope systems. For example, CESSAR-DC Section 8.2 defines interface requirements for the Offsite Power Systems. These sections and similar interface requirement sections for other systems were reviewed, and CDM Section 4.0 entries selected using the criteria discussed above.

14.3.5 CDM SECTION 5.0: SITE PARAMETERS

This section of the CDM defines the site parameters which were used as a basis for the design defined in the System 80+ standard plant design certification application. These entries respond to the 10 CFR 52.47(a) (1) (iii) requirement that the design certification documentation include site parameter information. The plant must be designed and built using the parameters in Section 5.0. Furthermore, it is intended that applicants referencing the System 80+ standard plant design certification demonstrate that these parameters for the selected site are within the certification envelope.

Site-specific external threats that relate to the acceptability of the design (and not to the acceptability of the site) are not considered site parameters and are addressed as interface requirements in the appropriate system entry.

Section 5.0 of the CDM does not include any ITAAC and is limited to defining site parameters. This is an appropriate approach because compliance of the site with these parameters must be demonstrated by a COL applicant prior to issuance of the license.

Selection Criteria - Section 2.0, Table 2.0-1 of CESSAR-DC provides the envelope of site design parameters used for the System 80+ standard plant design. The corresponding CDM Section 5.0 is based on using CESSAR-DC Table 2.0-1 in its entirety except as modified to meet the CDM content criteria previously discussed. For example, references in the CESSAR-DC table to specify Regulatory Guides have been deleted from the CDM table because of the guideline that the CDM does not contain direct references to codes and standards. Section 5 is limited to a tabular entry; no supporting text material is required.

14.3.6 ELEMENTS OF CESSAR-DC DESIGN MATERIAL INCORPORATED INTO THE CERTIFIED DESIGN MATERIAL

Tables 14.3-1 through 14.3-7 summarize the design material contained in CESSAR-DC that has been incorporated into the CDM in the areas of 1) Design Bases Accident Analysis, 2) Probabilistic Risk Assessment, 3) Shutdown Risk, 4) Severe Accident Analysis, 5) Flood Protection, 6) Fire Protection, and 7) Anticipated Transients Without Scram (ATWS). PRA assumptions incorporated into these tables encompass elements of the system design and assumptions that were expressly included in Tier 1 due to their importance. Both types of PRA assumptions are included for completeness, but are not distinguished in the tables.

CDM falling outside of the seven subject areas are intentionally not incorporated in these tables. However, the referenced CESSAR-DC sections may contain more information than just that encompassed by the these seven subject areas. Each table may also include design information (certified or non-certified) that is not directly related to the particular subject area. Further, the tables are not intended to include all system-specific CDM information that is provided in the CESSAR-DC system descriptions.

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overall assessment of the COLSS power distribution error. This is factored into the margin assessment as noted in Section 7.7.1.8.1.4.

7.7.1.8.1.3.4 Core Power Operating Limit Based on Peak Linear Heat Rate

The core power operating limit based on peak linear heat rate is calculated as a function of the core power distribution (F_q). The power level that results from this calculation corresponds to the LCO on linear heat rate margin.

7.7.1.8.1.3.5 Core Power Operating Limit Based on Margin to DNB

The core power operating limit based on margin to DNB is calculated as a function of the reactor coolant volumetric flowrate, the core power distribution, the maximum value of the four reactor coolant cold leg temperatures, and the Reactor Coolant System pressure. The CE-1 correlation is used in conjunction with an iterative scheme to compute the operating power limit. (See Section 4.4 for a detailed discussion of the CE-1 correlation). The power level that results from this calculation corresponds to the LCO on DNB margin.

7.7.1.8.1.4 Calculation and Measurement Uncertainties

The uncertainties in COLSS algorithms can be categorized as:

A. Uncertainties associated with the computation methods used to correlate the monitored variables to the calculated parameters.

B. The measurement uncertainties associated with the COLSS process instrumentation.

ROCS modified statistical combination of uncertainties

The COLSS is designed to accurately calculate power operating limits for normal core operating conditions. A large number of cases spanning the expected core operating conditions (1200 cases each at BOC, MOC, and EOC) are run using the COLSS ~~FORTRAN~~ simulation code and on ~~FLAIR~~, a 3-D reactor simulator code. These runs establish the modeling error between COLSS and the reactor simulator. This information, along with other appropriate data, such as CECOR errors and instrument errors, is used to determine COLSS power operating limit uncertainty factors which are then installed in the data base. The reactor simulator and error analysis codes are certified under the quality assurance program described in Chapter 17 (since they are also used for CPC analysis) while the COLSS uncertainty factors are reviewed independently. The uncertainty analysis methodology is documented in ~~CENPD-169~~ CEN-356 (Reference 1).

7.7.1.8.2 **NSSS Monitoring Programs**

The DPS application programs, exclusive of COLSS, that provide either a reactor monitoring or Plant Protection System monitoring function are described below:

A. CEA Position Monitoring Program

The DPS receives CEA positions from 2 sources, the CEACs (2 channels) and the CEDMCS (1 Channel). CEA position determination by these 2 separate sources is diverse. The CEAC utilizes reed switch position transmitters to sense CEA position while the CEDMCS senses the up and down movement of each drive mechanism to determine CEA position.

The CEA position, as obtained from the CEDMCS, is used directly as input to NSSS application programs. CEA positions determined by CEAC and CEDMCS are compared and validated to derive a validated CEA position for each rod. Differences in position, as determined by the diverse CEA position systems (CEDMCS and CEAC), are alarmed via DIAS. The validated CEA position information is used for display and data logging purposes.

B. CEA Trip Report Program

Upon detection of a reactor trip, a CEA trip processing program is activated within the CEDMCS which determines the rod drop time. This information is then sent to the DPS which compares them to the maximum allowable drop time and generates a report of the CEA trip behavior.

C. CEA Reassignment Program

Provisions to reassign individual CEAs to various CEA groups are provided to allow reconfiguration as would occur during a refueling outage.

D. CEA Exposure Accumulation Program

The CEA exposure accumulation program determines the approximate thermal megawatt hours of exposure for each CEA element based on average core power.

REFERENCES FOR SECTION 7.7

(1) "Ass~~essment of the~~ Accuracy of PWR ~~Operating Limits~~ as Determined by the Core ~~Operating Limit~~ Supervisory System," Co~~mbustion Engin~~eering, Inc., CENPD-169, ~~July 1975.~~

(2) "Overview of the Core Operating Limit Supervisory System," Combustion Engineering, Inc., CEN-312, Revision 01-P, November 1986.

(3) "Nuplex 80+ Software Program Manual," NPX80-IC-0101.0.

"Modified Statistical Combination of Uncertainties", CEN-356(V)-P-A, Rev. 01-P-A, Combustion Engineering, Inc., May 1988.

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APPENDIX B

LOWER MODE OPERATIONAL GUIDANCE

3.0 CONTENT OF LMOG APPENDIX

The content of this Appendix is generally consistent with the intent of the Safety Functions that are typically dominant during shutdown events. They are:

1. Reactivity Control - events that reduce boron concentration or cause CEA withdrawal.
2. RCS Inventory Control - events that drain the RCS or that cause loss of control of RCS inventory (such as during midloop operations).
3. RCS Heat Removal - events that cause loss of the shutdown cooling system capability.
4. Containment Integrity - events that cause radiological release directly out of an open containment, as during an outage, or indirectly through systems that interface with the RCS.

The particular events that challenge these safety functions may be somewhat different in detail than events initiated from the critical power modes. In the shutdown risk evaluations reported in CESSAR-DC Appendix 19.8A the shutdown specific topics are identified. A summary of the procedural guidance from Appendix 19.8A is given here in Table B-1. It lists seven topics for which procedural guidance related to shutdown operations is provided. For each topic, Table B-1 lists significant aspects that are addressed and also lists the relevant sections of Appendix 19.8A where there is additional information. These topics are expanded in the following sections of this LMOG.

To assure proper response to mitigate releases of radioactivity to the outside atmosphere, the COL applicant will:

a) Develop plant procedures to rapidly close the containment equipment hatch and other containment penetrations that are not closed from the control room, automatically closed by a containment isolation signal, automatically closed by an actuation signal, e.g. high radiation signal, or normally closed. The procedures would specify the personnel and equipment required (including prestaging) and the detailed steps to be taken.

b) Insure availability of instrumentation to monitor containment temperature, pressure and radiation during operations in Modes 5 and 6.

5.3.3 Isolation (closure of a containment isolation valve) in the non-operating SCS loop can reduce the possibility of an inadvertent draindown to the RCS.

5.3.4 Operations directly affecting the reactor vessel pressure boundary, i.e. In-core Instrumentation Seal Table evolutions, shall be prohibited during mid-loop operations.

5.3.5 OPERATIONAL PROCEDURES ARE ESTABLISHED TO RAPIDLY CLOSE SPECIFIED CONTAINMENT PENETRATIONS. PERSONNEL TRAINED IN THESE PROCEDURES ARE AVAILABLE TO EXECUTE THE PROCEDURES AS REQUIRED BY THE PLANT OPERATING MODE.

5.4.0 OPERAT[illegible] GUIDANCE

5.4.1 Verify RCS vent path established per Technical Specification (3.10.3).

5.4.2 Verify that the shutdown cooling/containment spray cross connection isolation valves are administratively closed.

5.4.3 Perform the RCS drain procedure to lower RCS level to the desired reduced inve[illegible]y elevation identified below:

Scheduled Maintenance Activity	RCS Elevation
S/G cold leg nozzle dams	[ ]
S/G hot leg nozzle dams RCP seal housing removal	[ ]
DVI nozzle 2A or 2B valve maintenance	[ ]

5.4.4 Monitor the following RCS/SCS system parameters during reduced inventory operations.

RCS core exit temperature	[List instruments]
SCS system flow rate	[ ]

5.5.0 ABNORMAL OPERATING CONDITIONS

5.5.1 Loss of shutdown cooling flow.

NOTE

There are a number of potential initiators that lead to the loss of shutdown cooling flow. The more probable initiators and the immediate actions to restore decay heat removal are discussed below. UPON LOSS OF SHUTDOWN COOLING, ACTIONS SHOULD BE TAKEN IMMEDIATELY TO INITIATE CONTAINMENT CLOSURE TO PRECLUDE RADIATION RELEASE TO THE OUTSIDE ATMOSPHERE.

A. Pump failure, i.e., bearing failure, motor failure, shaft breakage, etc.

Actions

1. Verify RCS level > minimum RCS level

2. Align the alternate SCS division, if required, for decay heat removal.

3. Start alternate division SCS system pump and verify decay heat removal capability.

4. Align the containment spray pump in the failed division for operation; hold system in standby.

5. Determine cause of SCS pump failure and determine most reliable means (division) of heat decay removal. Realign plant systems, if required, to support decay heat removal operation. If technical specification surveillance requirements/LCOs cannot be met, actions should be taken to raise RCS level to > elevation [117'-0"] as soon as possible.

B. SCS flow degradation due to vortexing

Amendment to Response 440.170

Amendment to Response 440.170:

As a result of a meeting held with the Staff on February 8-10, 1994, and phone conversations held on February 15 and 18, 1994, concerning the subject of rapid RCS drain down events in shutdown modes, the following amendment to this response has been prepared.

RISK ANALYSIS OF RAPID DRAINDOWN PATHS IN SHUTDOWN MODES

INTRODUCTION:

Seven major draindown paths were identified in the shutdown risk assessment (Appendix 19.8A). Most of the draindown paths and relevant valves are shown in Figure 1.

There were 2 major draindown paths identified in Table 2.12-2 of Appendix 19.8A. These are the Steam Generator (SG) manways and the SG nozzle dams. These paths are from the RCS directly and do not lead to a draindown to the bottom of the hot legs and therefore do not lead to loss of shutdown cooling. Five paths were identified in Table 2.12-3 of Appendix 19.8A. A review of the individual paths shows that one contains spring loaded safety valves (path #2), and three paths contain three or more MOVs which are to be locked closed (paths #4, #10, #13). The cross connect on the suction side of the CSS (path #6) is also addressed. The probability of having a draindown through these paths depends strongly on the procedures for entering shutdown cooling. Therefore the following section describes the assumed procedures for entering shutdown cooling.

ENTERING SHUTDOWN COOLING

The operator is entering shutdown cooling in Mode 4 with the containment closed. He has cooled the plant with the steam generators to 350°F and depressurized to 450 psi. He now will start the SCS trains because cooling through the SGs becomes less efficient and he wishes to quickly cool the plant down. He will be starting up both trains of SCS so that he can cool down the plant more rapidly. After the plant is cooled down, he will secure one of the SCS trains and leave it on standby. The following steps are assumed to be in written procedures with a checkoff list and are applied to both trains. The following description is for one train only.

1) Normal SCS Warmup
    - 1A In separate steps, with checkoff, the following valves are to be verified as closed: SI-340, SI-341, SI-657, SI-686, SI-300, SI-310, SI-688 and SI-314.
    - 1B Open SI-691 to establish recirculation flow in the SCS train.
    - 1C Verify SI-312 is open.
    - 1D Turn on SCS pump, check flow and pump characteristics. Procedural stop for abnormal pump characteristics.
    - 1E Open SI-651, SI-653, SI-655 aligning the suction side of

the SCS to the RCS.

1F Slowly open SI-601 while closing SI-691 to slowly heat the SCS train.

1G Ensure that SI-691 is closed and SI-601 is open after heatup.

At completion, the operator is on SCS.

2) Controlling Cooldown Rate

2A The operator must Slowly open SI-310 and closes SI-312 to control the amount of bypass around the shutdown cooling heat exchanger to cool the plant down at a desired rapid rate of 50 deg. F per hour (less than admin. limit of 75 deg. F/hr).

2B He will monitor the shutdown cooling exit temperature T-300 and primary coolant temperature to control cooldown rate.

After starting both SCS trains at a coolant temperature of 350°F and pressure of 450 psi, the operator will continue to cool and depressurize the plant. We estimate he will enter Mode 5 (defined as having coolant at 210°F) as an RCS pressure of 350 psi. He continues to cool to a RCS temperature of 135°F and 0 PSIG before opening the system for reduced inventory operation or vessel head removal. This time was estimated at 20 hrs (see Figure 19.8.3-3). The average conditions during this cooldown are 175 psi and 175°F.

QUARTERLY TESTING OF CSS AND SCS

Many components in the CSS trains and SCS trains are tested quarterly. Failure to restore the valves to their correct position after the testing could be a precursor to having a draindown event.

The CSS pump and MOV valves are tested quarterly. The suction line from the IRWST (SI-304) is normally open and discharge lines to the IRWST (SI-657, SI-686, and SI-300) would be opened. SI-657 and SI-686 are in draindown path #10. SI-300 is in drain down path #10, #4 and #13.

The SCS is also tested quarterly. Valve SI-340 is opened on the suction side. The valves in draindown path #13 (SI-314, SI-688, and SI-300) are opened to establish discharge flow.

The cross connect valve on the discharge side, SI-341, is also tested quarterly. This valve is not used for any normal function of starting SCS or CSS and is only used for cross connecting the two systems. This valve is in draindown path #10

These tests are on a safety system and require that the valve position be separately verified by another person before the test is complete. In addition, the valve positions are given in the control room and all but SI-300 are to be locked closed.

The NUPLEX 80 contains the COMAX, a computer aided testing system that compares the desired positions to the actual position for the valves with control room indication. This system verifies the valve test arrangement both before and after surveillance tests on safety systems. COMEX should identify to the operator, valves that were not restored after the above tests were performed. No credit was taken for this system in the risk analysis.

Adjust Boron Concentration

The operator might have to adjust the boron concentration in the SCS train. This should be required very infrequently because quarterly testing will keep the boron concentration at the IRWST concentration. The operator will align the SCS train to the IRWST and pump borated water through flow path #13. Again this is a potential sequence that might lead to a misaligned valve.

Path #2, LTOP Relief Valves

There are two spring loaded safety relief valves to protect the RCS and SCS from overpressure during Mode 5 and 6. Each valve is taken off of each SCS intake piping and is available whenever the SCS is aligned to the RCS. These valves are available even if the SCS train is on standby. If one of these valves fails to close, it can be isolated from the RCS by isolating the SCS train.

Spurious opening of a safety relief valve has rate of 5.0E-6/hr (Ref. EPRI-URD). For Mode 5, It was estimated that it would take 20 hours to cool the plant down in Mode 5 before opening the primary system. Assuming an 18 month refueling and one forced outage per year, the time exposed to this drain path is 33 hrs/yr (20 hrs/refuel * 0.6667 refueling/yr + 20 hrs/outage). We are assuming that the probability of spurious opening of the LTOP valves is negligible after depressurization. Since there are two LTOP valves, the draindown frequency is 3.3E-4/yr (5.0E-6/hr * 2 * 33 hr/y).

If an LTOP valve spuriously opened, the best estimate for drain down to the bottom of the hot leg would be 26 minutes. Boiloff to the active core would take another 19 minutes and heatup to core damage would take up to 20 minutes. The operator would have approximately 65 minutes to restore inventory or start a feed and bleed operation before CD. This is sufficient time to close the containment (i.e. greater than 1 hour). Given that there is a draindown, the operator has sufficient alarms to diagnose and take action before the onset of core damage (CD). The operator has low pressurizer level alarms, low RC level alarms, holdup tank alarms, containment temperature and radiation alarms and others. He can also take all necessary actions from the control room. In this analysis, core damage is defined as the onset of clad damage or reaching a temperature of 2200°F.

The operator has approximately 45 minutes to diagnose the transient and start some form of injection before uncovery and up to 20 minutes additional time before core damage. The operator would be assisted by the SRO and the shift supervisor for most of this time (after 5 minutes, S&G, T20-4, #3). The shift supervisor would assist with moderate dependency (MD, HEP = 0.14). The SRO is assumed to have high dependency but no credit was taken for his support. Also no credit is taken for the STA which could assist after 15 minutes. (S&G, T20-4, #3, and T20-17)

The operator's failure to respond in 30 minutes (Table value) is estimated at 0.001/d (S&G, T20-3, #4). Support from supervisor reduces this to HEP = 1.4E-4/d (0.001 * 0.14). If one took credit for the full hour, the HEP is reduced by an order of magnitude. In addition, the stuck open LTOP valve produces a bleed path for feed and bleed and returns the coolant indirectly to the IRWST. The core damage frequency is the product of the draindown frequency times the failure probability of the operators.

<u>CDF (path 2) = 4.62E-8/yr</u>.

PATH #13, FROM THE SCS DISCHARGE THROUGH THE FULL FLOW RETURN LINE

The operator had flow path #13 (through SI-314, SI-688, and SI-300) open for the quarterly flow tests. In very unusual conditions, he may have this path open for adjusting the boron concentration before using the SCS. Through a pre-existing maintenance error, the flow path could be left open.

Pre-existing maintenance error = 1.71E-4/d
- Pre-existing maint. error SI-314 = 8.73E-4/d (analysis of EFWS valve error in CESSAR-DC, p 19.5E-26)
- Pre-existing maint. error SI-688 = 0.14 (mild dep., S&G, T20-17, #3)
- Pre-existing maint. error SI-300 = 0.14 (above)

Walkdowns and control room scans = 0.14/d
- Walkdowns (0.52/d each of three valves, S&G, T20-27, #1) = 0.14/d
- Control Room Scanning = 1.0 (not quantified)

The most probable cause of the draindown is to have the draindown path left open and enter SCS. This would occur in Mode 4 while the containment would be closed.

Starting up the SCS Train (startup sequence is described on page 1):

1A: Check valve alignments, operator should discover the error here. HEP = 0.01 (failure to use check list at all, S&G, 20-6, #3)

1B: Open valve SI-691, No effect

1C: Check SCS heat exchanger bypass alignment, no effect.

1D: Start SCS pump for recirculation, Get low pressure alarm on P-300 (suction side of SCS pump), pump cavitation, current indicator is incorrect, Operator given Procedural Stop in written procedures. HEP = 0.025 (S&G, T20-2, #2, error per critical step in procedures)

1E: open suction side of SCS to RCS pressure causes a rapid draindown through the SCS pump to the IRWST.

The combination of the above events estimates the probability of having a rapid draindown event through path #13 during SCS startup, frequency = 6.0E-9/yr.

Recovery Actions:

1) The rapid draindown occurred during a change in state while the operator was opening valves, and the operator would tend to close the valves he was opening. This sequence is not a spurious failure.

2) Operator has the necessary alarms for the draindown and could activate SIS, or other makeup operations. The HEP for operator action would be less than that used for path #2, HEP = 4.62E-4/d, which would be used in this calculation.

The Core Damage Frequency is the product of the draindown frequency times the recovery actions (6.0E-9/yr * 4.62E-4/d)

<u>CDF (Path 13) = 2.77E-12/yr</u>

The above discussion addresses a draindown through path 13 caused by a pre-existing maintenance error. This sequence occurs while starting SCS. At the end of the refueling, this path is actually established to pump the coolant back to the IRWST from the refueling cavity. At this time the system is open, depressurized, and of lower decay heat. Draining down beyond the desired level is possible but would be detected by level instruments. If the draindown was continued to the bottom of the hot leg, the operator would have approximately two hours to make up inventory. This sequence has not been quantified because of the operator response time.

PATH 4, BACKFLOW THROUGH THE SCS RECIRCULATION LINE

Backflow through the recirculation line requires that valves SI-691, and flow path #13 be left open. It also requires the SCS pump be off. The risk of this path is contained in the risk of Path 13 which would be the actual draindown path before the SCS pump is secured.

PATH 10, THROUGH DISCHARGE SIDE CROSS CONNECT TO FLOW TEST PIPING

This path requires the misalignment of valves SI-341, SI-657, SI-686, and SI-300. All of these valves have position indication in the control room and all but SI-300 are locked closed. Valve SI-341 is not used for any normal or systems test procedure but is only used when aligning the CSS pump to substitute the SCS pump. The valve itself is tested quarterly. Valves SI-657, SI-686, and SI-300 are opened to flow test the CSS pump.

The risk of a draindown through this path while starting up the SCS train is identical to Path #13 since this path parallels Path #13. It does require the misalignment of one additional valve (SI-341) which is not normally used. This path would also cause SCS pump cavitation during startup, but before aligning to the RCS. The risk would be slightly less than Path #13 and could be conservatively assumed to be the same.

CDF (Path 10, Mode 4) = 2.77E-12/yr

Draindown through Path #10 is also possible when the CSS pump is used for DHR in Mode 5. This sequence would require:

1) Preexisting maintenance error on valves SI-657, SI-686, and SI-300 (HEP = 1.71E-4/d, from the path 13 analysis) and failure to detect with walkdowns and control room scans (HEP = 0.14/d) but SI-341 is correctly closed.

2) Failure to diagnose valve error on SCS startup (HEP = 0.01, step 1A of path 13).

3) Loss of DHR in Mode 5 during the 33 hrs/y of cooldown period (2.2E-3/d, BNL freq. SCS train 1).

4) Failure to start second SCS train (3E-2/d, Section 19.8)

6) Correctly opening valve SI-341 to use CSS pump (assumed successful).

These sequence of events leads to a rapid draindown through path 10 during cooldown in Mode 5. The frequency for this draindown is 1.58E-11/yr. The event is similar to the analysis of a draindown through path 6 where a total loss of DHR occurs with CSS maintenance.

Operator has the necessary alarms for the draindown and could activate SIS, or other makeup operations. The HEP for operator action would be similar to used for path #2, HEP = 4.62E-4/d, which would be used in this calculation. The CDF for this sequence is negligible.

PATH 6, CSS CROSS CONNECT ON THE SUCTION SIDE

The event tree (Figure 2) starts with the disassembly of one of the CSS trains. While the train is disassembled, the operating SCS train must fail and the second SCS train must fail to start. The operator then tries to cross connect the two systems. If the two systems are already cross connected, the CSS will not drain and disassembly would be impossible. A written step in the maintenance procedures will have the cross connect valve tagged out. In aligning the CSS to the SCS, a step in the written emergency procedure should be to verify that the CSS system is available. This is part of configuration management. Only if all of these events occur will you have a rapid draindown.

Given that a rapid draindown occurs, the operator must quickly diagnose the event and start injection. Failure to take action results in core damage (CD). The following paragraphs quantify the branches on the event tree:

CSS-D = 0.2/yr
The CSS must be disassembled for a rapid draindown through this cross connect to occur. It was assumed that this system is disassembled once every five years (the annual frequency is 0.2/yr). It is further assumed that the system is disassembled for 36 hours for maintenance (4.5 shifts).

L-DHR = 2.4E-3/D
The plant is being cooled by the operating SCS train. Failure to continue to run (Loss of DHR) during the 36 hour window that the CSS is disassembled is estimated at 2.4E-3/d. This is based on the hourly failure rate for loss of DHR of 1.79E-5/hr for Mode 5 from the BNL study.

F-SCS2 = 3E-2/d
The operator will try to start the second SCS train after the first SCS train fails. We are neglecting any recovery of the first train which is usually possible because many of the loss of DHR are operator errors. The operator has about two hours to react at this time in the sequence. Failure to start the second SCS train (F-SCS2) is dominated by failure of MOVs to open. This value is from Section 19.8 of CESSAR-DC.

F-TAG = 0.01/d

Failure of the maintenance group to tag out the cross connect valve SI-340 is estimated as 0.01/d. It is assumed that this is a step in the written maintenance procedures. This value is based on Swain & Guttmann (S&G, Table 20-6, #5) for failure to use valve change or restore list. If the valve control is correctly tagged out, the operator will not open the valve since it is the primary boundary.

F-CFG = 0.01/d

The operator will have written procedures to verify the availability of the CSS before aligning the CSS to the SCS for restoration of DHR. It is assumed the procedures is greater than 10 items long and the probability of omission (without checkoff) is 0.01/d (S&G, Table 20-7, #4).

F-LOCA = 0.01/d

The sequence of events described above leads to a rapid draindown with a frequency of occurrence of 1.44E-9/yr. It is assumed that the draindown occurs almost instantly and the coolant is drained to the bottom of the hot leg of the RCS piping. It is further assumed that the decay heat is high and boiling to the top of the active core takes 20 minutes. Heatup to the start of core damage (clad temperature of 2200°F) takes up to another 20 minutes. The operator has alarms on the pressurizer level, reactor vessel level, and sump drains. He should be able to identify the problem and take action. In this short time, the failure rate of the operator to take action was assumed to be 0.001/d. This is taken from S&G, Table 20-3, #4, failure to diagnose within 30 minutes.

Core Damage Frequency (CDF)

The frequency for rapid draindowns was estimated at 1.44E-9/yr. If CD actually occurs in 30 minutes, the is 1.44E-12/yr. If mist cooling of the core or lower decay heat extends the time to core damage by 10 minutes, F-LOCA improves by a factor of ten (S&G, T20-3, #4) and the CDF decreases to 1.44E-12/yr.

STEAM GENERATOR MANWAY

The SG manway was a identified as a major draindown pathway because of it's size. Draindown through the manway would leave the RCS coolant 8 inched above the centerline of the hot leg and would not cause loss of DHR. It is therefore not risk significant.

SG NOZZLE DAMS

Loss of the nozzle dams would cause the coolant to drain to the same elevation as the SG manway and not cause loss of DHR.

FIGURE 1
SIMPLIFIED SCHEMATIC FOR SCS AND CSS TRAIN 1

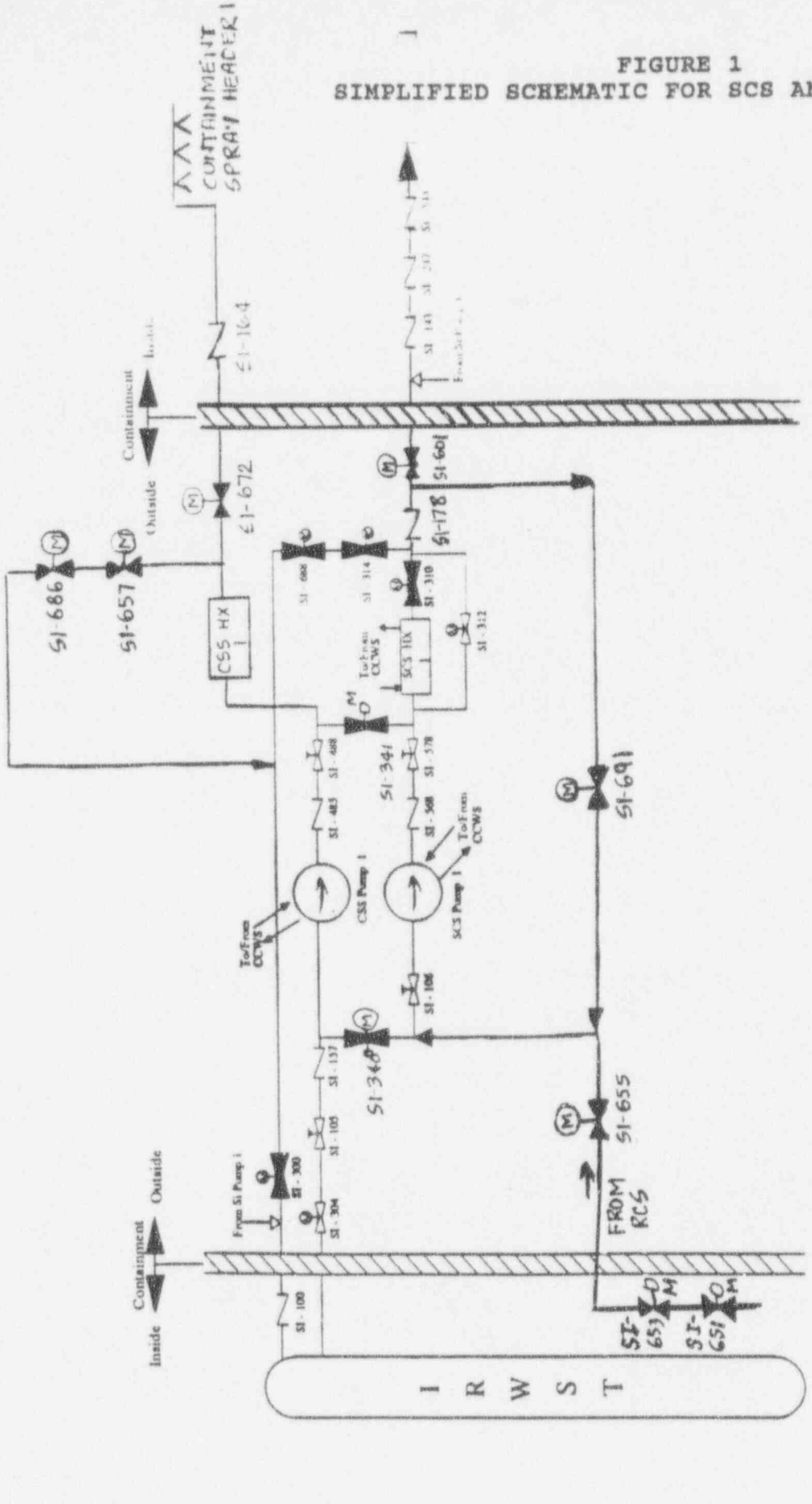
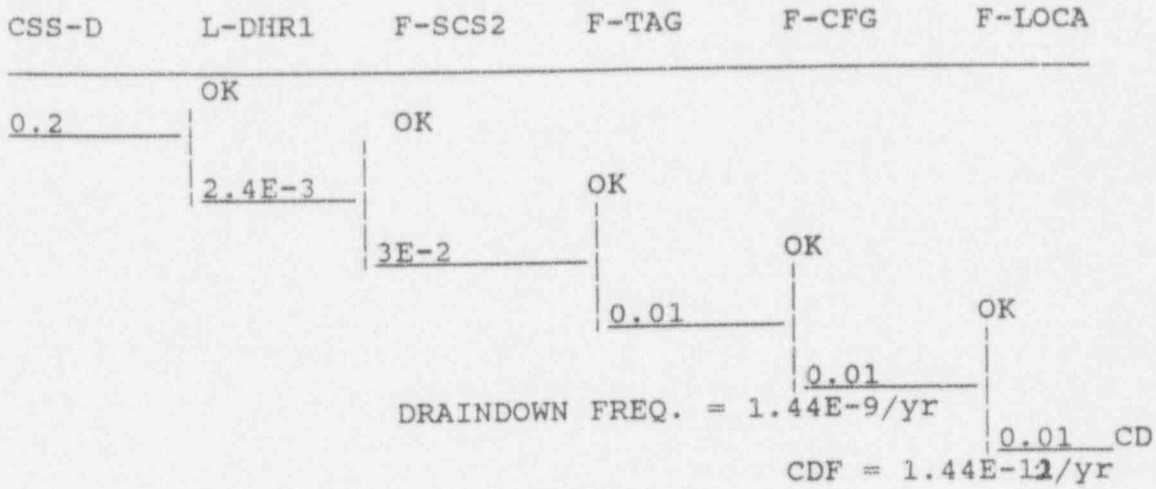


Figure 2
EVENT TREE FOR RAPID DRAINDOWN



Revisions to Technical Specification 3.10.5

3.10 REDUCED RCS INVENTORY OPERATIONS

3.10.5 Reduced RCS Inventory Operations - Containment Integrity

LCO 3.10.5 The containment building penetrations shall be in the following status:

a. The equipment hatch closed and held in place by [a minimum of four bolts,]

b. One door in each airlock closed,

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

1. Closed by an isolation valve, blind flange, manual valve, water, or equivalent; or

2. Exhausting through OPERABLE Reactor Building Containment Purge Exhaust System HEPA filters and charcoal absorbers, and is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: MODE 5 with REDUCED RCS INVENTORY

and

MODE 6 with REDUCED RCS INVENTORY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Restore containment penetration to required status.	[6 hours]
B. Required Action and Completion Time not met.	B.1 Restore RCS level to >[EL -117'0"].	[6 hours]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.10.5.1 Verify each required containment building penetration is in its required status.	[12 hours]
SR 3.10.5.2 Verify the Surveillance Requirements of SR 3.9.3.2 are met.	[18 months]

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TABLE 1.10-1 (Cont'd)

(Sheet 9 of 10)

COL LICENSE INFORMATION

Item Number	Subject	CESSAR-DC Section
14.2.4-1	Initial test program	14.2.4
14.2.6-1	Test records	14.2.6
14.2.9-1	Trial use of plant operating and emergency procedures	14.2.9
14.2.10-1	Initial fuel loading	14.2.10.1
14.2.11-1	Test program schedule	14.2.11
14.2.12.3-1	Scoping documents containing testing objectives and acceptance criteria	14.2.3
14.2.12.3-2	Documents listing plant conditions required during testing	14.2.3
14.2.12.3-3	Reconciliation methods for test conditions	14.2.4.3
14.2.12.3-4	Preoperational and startup test procedures	14.2.3.2
14.2.13-1	Security system and its test and acceptance criteria	14.2.7.5
15.3.10-1	Liquid tank failure	15.7.3.4
-	O-RAP description	17.3.9
17.3.1.2-1	O-RAP development and implementation	17.3.10
-	D-RAP	17.3.13
-	Validation of operating ensemble	18.9.3.2
18.9-1	Site-specific operating ensemble validation	18.9.3.2
19.1.2.2.2-1	Vulnerability of the intake structure due to tornado-generated debris	~~19.7.2.1.3~~ 19.15.1.3
19.1.2.2.3-1	Elements of the plant affecting the performance of systems in seismic events	19.7.5.3
19.1.2.2.6-1	Analysis using site-specific spectra	19.7.5.3

TABLE 1.10-1 (Cont'd)

(Sheet 10 of 10)

COL LICENSE INFORMATION

Item Number	Subject	CESSAR-DC Section
19.1.2.2.6-2	Site-specific examination of all external event hazards	~~19.7~~ 19.15.1 (see COL item 19.15.1-1)
19.1.2.2.6-3	Details and layout of the critical components and fire-suppression systems	~~19.7.3.1.6~~ 19.15.1
19.1.2.2.6-4	Potential internal flood source interaction and the details of the layout of the critical components	~~19.7.4.1~~ 19.15.1
19.1.2.2.6-5	Effects of the fire suppression systems on other systems	~~19.7.4.1~~ 19.15.1
20.1-1	Steam generator tube inservice inspection program	A-4
20.2-1	Reactor vessel supports material properties and 60-year neutron fluence	GSI 15(A)
20.2-2	Improving the reliability of open cycle service water systems	GSI 51(A)
20.2-3	Effects of fire protection systems actuation on safety-related equipment	Appendix A
20.2-4	Control room habitability	GSI 83(A)
20.2-5	Steam binding of auxiliary feedwater pumps	10.4.9.5.2
20.2-6	Piping and the use of combustible gases in vital areas	GSI 106(A)
20.2-7	Hydrogen control for large, dry PWR containments	Appendix A
20.2-8	Essential service water pump failures at multi-plant sites	Appendix A
20.2-9	Snubber operability assurance	3.9.3.4
20.2-10	Issue I.A.1.4	Appendix A
20.2-11	Issue I.C.9	Appendix A
20.3-1	Issues I.A.4.2 and II.J.3.1	Appendix A

Insert additional COL Items

Additional COL Items for Table 1.10-1 from PRA Insights

Item Number	Subject	CESSAR-DC Section
17.3.1-1	O-RAP requirement	17.3.1
19.7.5.3-1	Development of detailed seismic walkdown procedures	19.7.5.3
19.7.5.3-2	Comparison of as-built SSC HCLPFs to those assumed in the SMA	19.7.5.3
19.11.3.8-1	Calculation of specific flow rate for emergency containment spray pumping device.	19.11.3.8
19.11.3.8-2	Consideration of shielding requirements for local operator actions for the emergency containment spray backup system.	19.11.3.8
19.15.1-1	Update of PRA to include final design detail and site specific information	19.15.1
19.15..3.2-1	Fire brigade	19.15.3.2
19.15.6-1	List of risk significant SSCs for D-RAP and O-RAP	19.15.6
19.15.6-2	Consideration of risk important operator actions in developing procedures, training and human reliability related programs	19.15.6
19.15.6-3	Systems to address in severe accident management procedures	19.15.6
19.8.1.2-1	Establishment of administrative controls, outage management, procedures and training for shutdown operations.	19.8.1.2
19.8.1.2-2	Configuration control of fire and flood barriers during shutdown operations	19.8.1.2

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13.4-1	Reviews and audits	13.4

TABLE 1.10-1 (Cont'd)

(Sheet 8 of 10)

COL LICENSE INFORMATION

Item Number	Subject	CESSAR-DC Section
13.1-1	Organizational structure of the site operator	13.1
13.3-1	Site-specific emergency planning	13.3.2
13.3-2	Emergency planning support facilities	13.3.3.2
14.2.1-1	Startup administrative manual	14.2.1.1
14.2.2-1	Information on organization and staffing	14.2.2.1
14.2.3-1	Initial test procedures	14.2.3
14.2.4-1	Initial test program	14.2.4
14.2.6-1	Test records	14.2.6
14.2.9-1	Trial use of plant operating and emergency procedures	14.2.9
14.2.10-1	Initial fuel loading	14.2.10.1
14.2.11-1	Test program schedule	14.2.11
14.2.12.3-1	Scoping documents containing testing objectives and acceptance criteria	14.2.3.1
14.2.12.3-2	Documents listing plant conditions required during testing	14.2.3.2
14.2.12.3-3	Reconciliation methods for test conditions	14.2.4.3
14.2.12.3-4	Preoperational and startup test procedures	14.2.3.2
14.2.13-1	Security system and its test and acceptance criteria	13.6.1
15.3.10-1	Liquid tank failure	15.7.3.4
17.3.1.2-1	O-RAP development and implementation	17.3.10
19.1.2.2.2-1	Vulnerability of the intake structure due to tornado-generated debris	19.7.2.1.3

13.5-1 Site-specific plant operating procedures 13.5.1-2
18.9-1 Site-specific operating ensemble validation 18.9.3.2

TABLE 1.10-1 (Cont'd)

(Sheet 8 of 10)

COL LICENSE INFORMATION

Item Number	Subject	CESSAR-DC Section
13.1-1	Organizational structure of the site operator	13.1
13.3-1	Site-specific emergency planning	13.3.2
13.3-2	Emergency planning support facilities	13.3.3.2
14.2.1-1	Startup administrative manual	14.2.1.1
14.2.2-1	Information on organization and staffing	14.2.2.1
14.2.3-1	Initial test procedures	14.2.3
14.2.4-1	Initial test program	14.2.4
14.2.6-1	Test records	14.2.6
14.2.9-1	Trial use of plant operating and emergency procedures	14.2.9
14.2.10-1	Initial fuel loading	14.2.10.1
14.2.11-1	Test program schedule	14.2.11
14.2.12.3-1	Scoping documents containing testing objectives and acceptance criteria	14.2.3.1
14.2.12.3-2	Documents listing plant conditions required during testing	14.2.3.2
14.2.12.3-3	Reconciliation methods for test conditions	14.2.4.3
14.2.12.3-4	Preoperational and startup test procedures	14.2.3.2
14.2.13-1	Security system and its test and acceptance criteria	14.2.3.3
15.3.10-1	Liquid tank failure	15.7.3.4
17.3.1.2-1	O-RAP development and implementation	17.3.10
19.1.2.2.2-1	Vulnerability of the intake structure due to tornado-generated debris	19.7.2.1.3

ATTACHMENT 10

on the ratio of the DB to RLE and the criteria outlined in EPRI NP-6041-SL[176], the HCLPF was determined. Ceneric Equipment Ruggedness Spectra (GERS) were used where applicable.

Solid state switching devices and electromechanical relays will be used in the NUPLEX 80+ protection and control systems. Solid state switching devices are inherently immune to mechanical switching discontinuities such as contact chatter. Robust electromechanical relays are selected for NUPLEX 80+ applications such that inherent mechanical contact chatter is within the requisite system performance criteria. Therefore, contact chatter has no effect on system operation and was, therefore, not included in the seismic analysis. The COL must confirm the use of seismically robust electromechanical relays in the engineered safety features actuation and control systems. [COL Item 19-20]

Similarly, for equipment qualified by analysis, the design basis parameters, including damping and anchorage safety factors as well as response spectra, were assessed to determine seismic margins. For example, a seismic margin exists for an equipment item that is seismically qualified using 3% damping, if a higher 5% damping is permitted in the CDFM approach. Margins for each of the significant parameters were determined, and considered with inelastic energy absorption capability for ductile failure modes, to calculate HCLPFs. Seismic fragilities for equipment items that are inherently rugged were based on available industry data.

The NSSS components that were reviewed include:

A. Reactor Vessel
B. Steam Generators
C. Reactor Coolant Pumps
D. Reactor Internals
E. Pressurizer
F. NSSS Piping

These NSSS components were assessed by reviewing the various design basis parameters to identify seismic margins. The stress margins were determined by comparing the loads from the System 80+ RCS analysis and those in existing stress reports. Load combinations in accordance with Section 6 of EPRI NP-6041-SL[176] were evaluated. The HCLPF values were calculated using the CDFM approach and the equations discussed above.

Table 19.7.5.1-1 lists all components that are in the seismic models and presents either the failure rates for independent failures or the component or structure HCLPF value for seismic failures. Table 19.7.5.1-2 lists the structural fragilities calculated for the System 80+ structures and the NSSS component fragilities. All HCLPF values are presented in terms of peak ground acceleration.

19.7.5.3 Seismic Margins Results and Insights

Table 19.7.5.3-1 presents the HCLPFs for System 80+ by sequence. The data on this table is sorted in ascending order on the HCLPF values in the second column. As can be seen from this table, the HCLPF for System 80+ is 0.73g. The dominant contributor to the plant HCLPF is seismically induced gross structural failure due to a seismically induced failure/overturning of the containment vessel which is assumed to lead directly to core damage and containment failure. The second dominant contributor to the plant HCLPF, with a HCLPF of 0.86g, is a seismically induced LOCA in excess of ECCS capacity caused by a seismically induced failure of the RCP supports.

There are three sequences where the contribution to the plant HCLPF due to "Mixed Cutsets" is potentially significant. (NOTE: A "Mixed Cutset" is a core damage cutset which contains both seismic failures and random or independent failures.) These three sequences are SEIS-SBO, EQA-15, and EQA-9.

The sequence SEIS-SBO is a seismically induced station blackout. The sequence is initiated by a seismically induced loss of site power, which has a HCLPF value of 0.12g. Failures of the two diesel generators and the standby combustion turbine independent of the seismic event will result in a station blackout. The station batteries and the turbine driven EFW pumps can be used to provide emergency feedwater flow for approximately 8 hours. At this time, the batteries will be depleted and the turbine driven EFW pumps will fail due to loss of control power. This results in core damage. It is assumed that offsite power can not be restored within 24 hours of a seismic event which results in loss of offsite power. Thus, the emergency diesel generators and the standby combustion turbine are important to maintaining the safety of the plant following a seismic event. The diesel generators and the structures in which they are housed are Seismic Category I and are designed to have significant margins over the design basis earthquake. It is also highly desirable that the standby combustion turbine be available to back up the diesel generators for seismic events on the order of the design basis earthquake of 0.36g. As stated in Chapter 6, paragraph 4.6.4.3.5 of volume II of the EPRI Utility Requirements Document[7], practical measures can be taken to exploit the intrinsic seismic ruggedness of the combustion turbine to provide a median seismic capacity of 1.0g. The ARSAP letter report, "Combustion Turbine Generator Seismic Evaluation Guidance WBS 4.1"[219] provides specific guidance on considerations for ensuring the seismic strength of the combustion turbine and the structure in which it is housed. ⟨Insert 19.7.5.3-0⟩

Sequence EQA-15 is a seismically induced ATWS early in core life when the MTC is greater than -0.3. The peak RCS pressure for an ATWS with an MTC greater than -0.3 would exceed 3200 psia which would result in failure of the safety injection system check

Insert 19.7.5.3-0

The structure that houses the combustion turbine must either have a HCLPF value of at least 0.36g, which is consistent with the HCLPF value assumed for the combustion turbine in the seismic margin analysis, or be shown to fail in such a manner that it will not reduce the HCLPF value of the combustion turbine below 0.36g.

[COL ITem 19-8]

capability sequence. The results of these sensitivity analyses are summarized in Table 19.7.5.3-2. This table defines the component or component group for which the HCLPF is being changed, presents the base component HCLPF and the changed component HCLPF, and presents the new "Min Sequence" HCLPF and the new plant HCLPF. These sensitivity analyses basically confirmed the intuitive importance of the component HCLPFs. That is, if the HCLPF for a component that appears as a dominant contributor to a sequence HCLPF is decreased below the plant or "Min Sequence" HCLPF values, the plant and/or "Min Sequence" HCLPF value will decrease to the lower value. On the other hand, if the HCLPF for that component or group of components is increased, the plant and "Min Sequence" HCLPF values do not change. [COL ITem 19-9]

None of the seismic core damage sequences had a HCLPF less than two times the design basis earthquake. In addition, the dominant sequence is a seismically induced failure of the containment vessel which leads to core damage and containment failure. Therefore, no additional containment isolation failure analyses were performed.

Insert 19.7.5.3-1

seismic

including development of detailed procedures,

The COL applicant will be required to verify that key assumptions for structures, systems and components considered in the SMA are valid for the as-built plant conditions. The verification process will include a ~~plant~~ walkdown, to assure that proper anchorage for equipment has been provided and that the potential for seismic spatial system interaction does not exist. Also, if equipment is qualified for site-specific requirements (Appendix 3.9A, Section 1.4.3.2.1.2, Option 4), the impact on the SMA HCLPF will be evaluated.

Insert 19.7.5.3-2

In Section 19.1.2.2.3 (page 19-149) of the Draft Safety Evaluation Report for CESSAR-DC, it stated:

> "As the seismic analysis is being redone, there is a need to augment the internal events model to the extent possible, by explicit inclusion of structural and other passive failures that were excluded from the internal events model."

This is COL action item ~~19.1.2.2.3-1.~~ 19-2 As part of the model development for the Seismic Margins Analysis, the internal events model was updated to include structural and passive failures. This completes COL action item ~~19.1.2.2.3-1.~~ 19-2

ensure that as-built conditions conform to the assumptions used in the SMA and to

Deviations from assumptions will be evaluated to determine if vulnerabilities have been introduced.

Insert 19.7.5.3-3

Insert 19.7.5.3-1

This will include evaluation of HCLPF values for structures which house non-safety related equipment relied upon in the SMA evaluations such as the combustion gas turbine.

Insert 19.7.5.3-2

The details for verifying that key assumptions for structures, systems, and components considered in the SMA are valid for as-built plant conditions should be modeled after approved NRC SMA procedures and should incorporate insights from SMAs conducted at operating plants. The verification process is expected to consist of the following steps:

1. Preparation for Plant Walkdown

2. Plant Seismic Logic Model Walkdown

3. Assessment of As-Built SMA HCLPF Values

4. Seismic Plant Walkdown

5. Validation of Plant Level HCLPF Calculations

These steps will ensure that as-built plant design characteristics are evaluated; critical component, structural, and sequence HCLPF values are reviewed; and deviations from design assumptions and vulnerabilities which could reduce the plant level HCLPF value below 0.5g are exposed.

Insert 19.7.5.3-3

The Draft Safety Analysis Report for CESSAR-DC ~~is~~ contained ~~two~~ an additional COL item~~s~~, 19.1.2.2.6-1, (Item 19-3 in Table 1.10-1) which ~~required~~ stated that the COL applicant ~~to~~ should factor site-specific spectra into the analysis and verify the layout and anchorage of critical components. ~~and [illegible]~~ This COL item is covered by the SMA and the seismic walkdown to be performed by the COL applicant as discussed above.

TABLE 19.7.5.4-2 STRUCTURE AND MAJOR NSSS COMPONENT HCLPFs FOR ROCK AND SOIL SITES						
COMPONENT/ STRUCTURE	ROCK HCLPF	B1 SOIL HCLPF	B1.5 SOIL HCLPF	B2 SOIL HCLPF	B3.5 SOIL HCLPF	B4 SOIL HCLPF
Interior Structure*	1.08g	1.08g	1.08g	1.08g	1.08g	1.08g
Fuel Building*	1.35g	1.35g	1.35g	1.35g	1.35g	1.35g
CVCS*	1.01g	1.01g	1.01g	1.01g	1.01g	1.01g
Diesel Generator 1 or 2*	0.89g	0.89g	0.89g	0.89g	0.89g	0.89g
EFW Storage Tank 1 or 2*	0.89g	0.89g	0.89g	0.89g	0.89g	0.89g
Control Room Area*	1.12g	1.12g	1.12g	1.12g	1.12g	1.12g
Shield Building*	1.25g	1.25g	1.25g	1.25g	1.25g	1.25g
Containment Vessel*	0.73g	0.73g	0.73g	0.73g	0.73g	0.73g
Service Water Pump Building	1.00g	1.50g	1.50g	1.50 ~~1.10g~~	1.10 ~~1.00g~~	1.10 ~~1.00g~~
CCW Heat Exchanger ~~Service water Pump~~ Building - Sliding	1.00g	1.50g	1.50g	1.50 ~~1.10g~~	1.10 ~~1.00g~~	1.10 ~~1.00g~~
Nuclear Island - Sliding	1.00 ~~0.80g~~	1.30g	1.50g	0.90g	1.60g	1.20g
CEDMs (Rock governs except for B1)	1.35g	1.01g	1.35g	1.35g	1.35g	1.35g
RCP/Supports	0.86g	0.90g	1.49g	2.58g	1.89g	0.91g
Reactor Vessel/Supports	1.14g	1.87g	1.75g	3.20g	1.72g	1.13g
Reactor Internals - Spacer Grid*	0.75g	0.75g	0.75g	0.75g	0.75g	0.75g
Reactor Internals - Fuel Assy*	0.87g	0.87g	0.87g	0.87g	0.87g	0.87g

D. The concept of defense in depth applies to shutdown modes as well as Mode 1. The more ways that the operator can maintain coolant inventory and remove decay heat, the lower the risk. The presence of SIS capability in shutdown is an example of added defense in depth.

E. The ability of the operator to be able to align the SCS train for makeup of inventory or to use for a feed and bleed operation is important for defense in depth.

F. There are two trains of SCS and it is important that the COL applicant maintain a configuration management system for maintenance activities on the SCS and its support systems. Configuration control is important because all plant risks, all accidents and incidents, and all accident precursors arise because of critical configurations which have occurred. If configurations were managed so that critical, high-risk configurations did not occur, then the risks would be small and accidents or incidents would occur rarely. Table 19.8.1-4 (developed from the more extended dependency Table 19.6.1-1) is an example of the systems that support each SCS train. The COL applicant should identify the systems, structures and components (SSCs) that support DHR (as well as other safety functions). The COL applicant should consider the overall effect of removing SSCs identified above from service on the DHR safety function. The COL applicant should limit normal maintenance on combinations of equipment so that an additional single or common cause failure would not cause total loss of DHR. A configuration management system should help to insure the availability of the standby SCS.

G. If one train is lost because of fire, flood, or random component failure, it is important that the other train have the highest possible availability. The COL applicant should develop procedures and a configuration management strategy to handle the period of time when one of the two DHR paths is unavailable. In this case (a technical specification violation) the operator should suspend the maintenance and testing activities on equipment that support the operating SCS train. Given failure of one train, the operator should restore any systems that support the other train and are out for maintenance.

<Insert 19.8.1.2A here>

Loss of DHR Insights

A. Reduced inventory is the most critical operation. The operator should be aware of this and plant activities should be scheduled accordingly. Use of nozzle dams is encouraged as a method of limiting the time spent in this mode.

Insert 19.8.1.2A

H. During plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures and operator knowledge of plant configuration. The COL applicant should develop the appropriate administrative controls procedures and operator training for shutdown operations. (See also insights F and G above.) [COL Item ~~19.8.1.2-1~~]
19-17

I. During plant shutdown operation, the integrity of fire and flood barriers between areas in the same division, such as quadrants, where systems comprising the alternate shut down are located should be maintained. The COL applicant should incorporate in its configuration control program a requirement that, during modes 4, 5, and 6, the water tight flood doors and fire doors will be maintained closed on at least one quadrant within the subsphere (containing either an SCS or CSS pump) to help prevent common-mode failures from internal floods or fires. The SCS or CSS pump in this quadrant shall be operable. If the flood or fire doors to this quadrant must be opened for reasons other than to permit normal access a fire watch will be established for the affected door. [COL Item ~~19.8.1.2-2~~]
19-18

ground release to the subsoil and eventually, any underlying aquifer. This is discussed in more detail in Section 19.11.4.2. For this case, the releases are assumed to occur at ground level.

For isolation failures involving steam generator tube ruptures with stuck open steam relief valves, the releases occur in the main steam valve room. These releases are assumed to pass to the environment through the roof of the main steam valve room at 156+0 ft. This is 64 feet or 19.7 meters above grade. The releases due to other isolation failures are assumed to be released to the environment at about the level of the equipment and personnel hatchs at 146+0 ft. This is 54 feet or 16.6 meters above grade.

The isotopic content of the release for each release class was calculated using S80SOR, a version of the ZISOR code modified to reflect System 80+ design features. Calculation of the source terms and the S80SOR code are discussed in Section 19.11.4.3 and Appendix J to Section 19.11. One or more representative PDSs were selected for each release class, and a S80SOR run was made to calculate source term isotopic content for the specific Release class for that PDS. The source term isotopic content used for the release class was the weighted average of the source terms for the PDSs selected to characterize the release class.

General values for the time of core damage and the time of containment failure were selected based on the general definition of the core damage class and the release class. For early core damage sequences, the onset of core damage was assumed to occur at approximately 4 hours on the average with vessel failure approximately one hour later. For mid core damage sequences, with the exception of Station Blackout with battery depletion cases, the onset of core damage was assumed to occur at approximately 16 hours on the average with vessel failure approximately 1 hour later. For station blackout with battery depletion cases, the onset of core damage was assumed to occur at about 10 hours with vessel failure 1 hour later. For late core damage sequences, core damage occurs at some time greater than 24 hours. For this analysis, the core damage time was set to 24 hours for these cases. As with the other core damage time classes, vessel failure was assumed to occur 1 hour later.

Early containment failures were assumed to occur within 1 hour of vessel failure. Late containment failures due to late hydrogen burns were assumed to occur within 6 to 8 hours of vessel failure. Late containment failures due to overpressure failure were assumed to occur 65 hours after vessel failure. Basemat melt through failures were very conservatively also assumed to occur 65 hours after vessel failure. Actual basemat meltthroughs would take on the order of a week or more. All isolation failures were assumed to occur at time zero.

← Insert 19.12.3.1-1 here >

Insert 19.12.3-1

The timing of basemat melt-through was based on the minimum time required to for the corium to erode a dry reactor cavity basemat into the free space of the subsphere. Penetration of melt into the subsphere requires lateral erosion of approximately 6 feet of concrete. In actuality rapid lateral erosion into the basemat is unlikely. However, because the lateral erosion scenario was considered to be dominant from the viewpoint of above ground fission product releases this timing was used to characterize the release. Alternate definitions of release time considered were the time to penetration of the lower steel shell and the time for full penetration of the basemat. The releases associated with shell penetration were not considered to result in significant fission product releases and thus were lumped into the 6 foot penetration time used to represent the subsphere. The time to penetrate the full basemat was estimated to be about 8 days and therefore is not considered radiologically significant.

Containment overtemperature failures were not considered radiologically significant. The System 80+ design requires leak resistant penetration designs and utilization of sealants with high temperature stability. Analyses suggest the limiting containment temperatures to be in the range of 350 to 450 F. A review of containment penetrations (see Section 19.11.4.4) indicates that under these conditions typical containment penetrations (representative of those expected to be used in System 80+) will maintain leak tightness for several days or more following the severe accident.

unavailable due to depletion of the IRWST inventory. Releases would be via the unisolated ruptured steam generator.

The releases for this release class were assumed to start at the time of core damage at 25 hours and last for 24 hours. The release to environment was assumed to occur at an elevation of 19.7 meters above grade.

Release class RC4.18L covers releases associated with a containment isolation failure with vaporization releases and revaporization releases for sequences in which core damage occurs after 24 hours. In-vessel fission product scrubbing is successful for release class RC4.18L, but scrubbing of the vaporization and revaporization releases is not successful. The dominant PDS for this release class is also PDS194. For release class RC4.18L, scrubbing of the in-vessel fission product releases via the inventory in the SG is credited. S80SOR, a modified version of ZISOR, is used to calculate release fractions for the various release classes. S80SOR does not credit in-vessel fission product scrubbing via SG inventory. Thus, the releases calculated for RC4.18L are the same as those for RC4.36L. Therefore, RC4.18L is combined with RC4.36L. Release class RC4.18E is also combined with release class RC4.36L. The release frequency for RC4.18L is 1.54E-08, and the release frequency for release class RC4.18E is 2.06E-11. Therefore, the total release frequency for release class RC4.36L is 3.08E-08.

19.12.3.2.5 Release Class RC5.1E

Release class RC5.1 covers releases associated with a containment bypass failure with vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release but with the source term attenuated due to deposition in the auxiliary building. The cumulative release frequency for this release class is 5.10E-10 per year.

This release class was characterized by a failure of the check and isolation valves in one SCS line resulting in a catastrophic failure of this line outside containment (Interfacing System LOCA). Safety injection was successful. However, the primary system inventory and the IRWST inventory is discharged outside of containment. Core failure was assumed to occur at 2 hours and vessel failure at 3 hours. Containment spray is unavailable and the cavity is dry due to depletion of the IRWST inventory. The release path is through the broken SCS line into the subsphere region of the auxiliary building at elevation +50 ft. ~~If the flood doors are closed and do not fail, the subsphere shadow region will be flooded to a depth of about 9 feet. If the flood doors fail or are not closed, the affected quadrant at the 50+0 foot level will be flooded to depth of about 4 feet. In the former case, the fission products from the RCS are released under about 8 feet of water and thus are subject to substantial scrubbing. In addition, the water will cool the SCS piping, thus promoting deposition of the fission products in the SCS piping. In the latter case, the~~

Insert 19.12.3.2.5.1 h...

Insert 19.12.3.2.5-1

If the flood doors for the subsphere region Flood Zone containing the affected SCS line are closed and do not fail, the Flood zone would be completely flooded. However, because there is no steam/pressure release path if these doors remain closed, the flood doors for the subsphere region flood zone containing the affected SCS line are assumed to fail. This will lead to flooding of the adjacent flood zone. The adjacent flood zone,does have pathways for steam/pressure release so no other flood doors are assumed to fail. If the failed SCS line is in division 1, the affected flood zones will be flooded to a depth of approximately 5.6 feet. If the failed SCS line is in division 2, the affected flood zones would be flooded to a depth of approximately 6.4 feet.

~~fission products are released under less than 4 feet of water and are subject to less scrubbing. In addition, there is less deposition in the SCS piping~~. After release in the subsphere region, the fission products must then be transported to the stairwells or elevator shafts and passed upward through the auxiliary building until it eventually reaches a release point to the environment. For this case, the fission products are assumed to be released to the environment at ground level. Releases last for 24 hours following core damage at 2 hours.

isolation is assumed to occur or for another case where an isolation failure is assumed to occur, but not both. To resolve this issue, PDS181, PDS184, PDS193, PDS196, PDS218, and PDS220 were partitioned into two parts. The first partition represented the case where no isolation failure occurred, and the second partition represented the case where an isolation failure is assumed to have occurred. To assess the impact of the partitioning of the ruptured steam generator isolation failure on the various releases classes, a sensitivity analysis was performed. In this sensitivity analysis, for the case where no isolation failure occurred the partitioning was changed from 96% to 80%, and for the case were isolation failure was assumed the partitioning was changed from 4% to 20%. This assumes that the ruptured steam generator is likely to be isolated. As a result, the conditional probability for containment isolation failure releases is expected to increase.

The sensitivity results for this case are presented in Table 19.14.1-4 as case 10. The results show that the most affected containment failure mode would be the containment bypass releases (RC4). The conditional probability for containment bypass releases increased from 0.024 to 0.047. The containment bypass releases would increase by an approximate factor of 2. The conditional probability for early containment failure releases (RC3) would remain unchanged while both the intact containment releases (RC1) and the late containment failure releases (RC2) would decrease by a small amount, from 0.887 to 0.865 and 0.079 to 0.077, respectively. The results of this sensitivity analysis imply that the containment bypass releases are sensitive to the manner in which the six PDSs are partitioned to address isolation failure of the ruptured steam generator.

INSERT New Section 19.14.1.11

NEW SECTION

19.14.1.11 Use of Emergency Containment Spray Backup System

The use of the Emergency Containment Spray Backup System is credited as one of three ways of recovering the containment heat removal function in the long term. (See NCHRRECOV, section 19.12.2.2.7.1.1.1.2) The other ways of recovering the containment heat removal function were recovery of off-site power for sequence cutsets where containment heat removal was lost due to loss of offsite power, and operator initiation of containment spray for those sequence cutsets in which the containment spray system was unavailable due to failure of the operators to initiate containment spray. As discussed in section 19.12.2.2.7.1.1.1.2, for PDSs in which the containment sprays were unavailable early in a sequence, each cutset was evaluated to identify appropriate recovery actions from the above three recovery actions. A non-recovery factor was then determined for each cutset. An overall nonrecovery factor (NCHRRECOV) was then calculated for the PDS using a weighted average of the individual cutset non-recovery factors.

For this sensitivity analysis, the nonrecovery factor for the backup containment spray system was set to 1.0, that is, it was assumed that the backup containment spray system was always unavailable. The value for NCHRRECOV was then recalculated for each PDS in which the containment sprays were unavailable early in a sequence. The CET was then requantified for all PDSs. The results of this requantification are provided in table 19.14.1-1 as case 11. These results show that if the Emergency Containment Spray Backup System is not credited as a means of recovering containment heat removal in the long term, there is an increase in the conditional probability of a late containment overpressure failure with a corresponding decrease of the probability of the containment remaining intact in the long term.

TABLE 19.14.1-1 (Cont'd)

(Sheet 2 of 2)

SUMMARY OF CONTAINMENT RESPONSE SENSITIVITY ANALYSIS
RESULTS FOR SYSTEM 80+

CASE NO.	DESCRIPTION	MODELED AS	CONDITIONAL PROBABILITY OF RELEASE CLASS				
			RC1	RC2	RC3	RC4	RC5
7A	THE RCS IS NOT DEPRESSURIZED BY THE SDS FOR CORE DAMAGE SEQUENCES THAT INVOLVE CYCLING OF RELIEF VALVES	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 1.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.887	0.079	0.011	0.024	0.0
7B	THE RCS IS DEPRESSURIZED BY THE SDS FOR CORE DAMAGE SEQUENCES THAT INVOLVE CYCLING OF RELIEF VALVES	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 0.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.887	0.079	0.011	0.024	0.0
7C	THE RCS IS NOT DEPRESSURIZED BY THE SDS FOR MEDIUM AND HIGH RCS PRESSURE CORE DAMAGE SEQUENCES	CHANGE PROBABILITY OF "NOSDSDP" FROM NON-ZERO VALUES TO 1.0.	0.887	0.079	0.011	0.024	0.0
8	CONTAINMENT IS LESS LIKELY TO BE ISOLATED	CHANGE PROBABILITY OF "ISOL" FROM 2.1E-03 TO 1.0E-02.	0.881	0.078	0.011	0.030	0.0
9	OPERATOR IS LESS LIKELY TO TURN H_2 IGNITORS ON	INCREASE PROBABILITY OF "OPIGNITOFF" BY ORDER OF MAGNITUDE (FROM 3.0E-2 TO 3.0E-1)	0.883	0.079	0.014	0.024	0.0
10	ISOLATION OF RUPTURED STEAM GENERATOR	CHANGE PARTITIONING OF PPS181, PDS184, PDS193, PDS196, PDS218, AND PDS220 FROM 96/4 TO 80/20	0.865	0.077	0.011	0.047	0.0
11	Emergency Containment Spray Backup System Not Available ~~for~~ To Support Recovery of Containment Heat Removal	Non-recovery factor for Emergency Containment Spray Backup System set to 1.0 in Calculation of NCHRRECOV for PDSs with containment spray initially unavailable.	.862	.103	.011	.024	0.0

Section 19.15.1. Insights about the System 80+ design gained from the internal events risk profile and the external events risk profile are summarized in Sections 19.15.2 and 19.15.3, respectively. Shutdown and low-power operation are included as part of the System 80+ PRA, and the insights gained from the risk associated with these modes of operation are summarized in Section 19.15.4. The use of PRA in the design process is summarized in Section 19.15.5. The use of PRA results and insights to support certification and followup activities is summarized in Section 19.15.~~6~~7. Significant PRA-based safety insights for the System 80+ design are provided in Table 19.15-1.

< INSert 19.15.1A >

Insert 19.15.1A

During the detailed design phase for System 80+, site specific information and system design details will become available. The COL applicant should update the PRA using the final design information and site specific information. As deemed necessary, the update should include the shutdown risk evaluation and the internal fire and flood evaluation. Based on site specific information, the COL applicant should also re-evaluate the qualitative screening of external events. If any site specific vulnerabilities are found, the applicable external event(s) should be included in the updated PRA. [COL Item ~~19.15.1-1~~ 19-12]

In updating the internal fire evaluation, the COL applicant should verify the details and layout of critical components and the fire suppression systems. [COL Item ~~19.1.2.2.6-3~~ 19-5] The applicant should also evaluate the potential effect of the fire suppression systems on the behavior of other systems. [COL Item ~~19.1.2.2.6-5~~ 19-7]

In updating the internal flood evaluation, the COL should evaluate the interaction of the potential internal flood sources and the details of the layout of the critical components. [COL Item ~~19.1.2.2.6-4~~ 19-6]

TABLE 19.15-1

(Sheet 1 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

Conforms to INSIGHT	DISPOSITION
1. The COL applicant should perform a seismic walkdown to ensure that the as-built plant ~~matches~~ the assumptions in the System 80 + PRA -based seismic margins analysis and to assure that seismic spatial systems interactions do not exist. <INSERT 19.15-1A here>	~~COL ITEM~~ <CESSAR-DC 19.7.5.3
2. ABB-CE will maintain a list of the SSC HCLPF values used in the System 80+ Seismic Margins Assessment in the D-RAP. <Insert 19.15-2B here>	D-RAP
3. ABB-CE will maintain a list of risk significant SSCs in the D-RAP	D-RAP
4. The COL will maintain an O-RAP based on the system reliability information derived from the PRA and other sources. <Insert 19.15-4A Here>	COL ITEM ~~19.3~~ 17.7 (CESSAR-DC section 17.3.1)
5. 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 service water and component cooling water structures, prevents fires and floods from propagating from one division to the other.	Addressed in all safety-related structures and systems.
There are no doors or passageways connecting the divisions of safety-related equipment up to elevation 70+0.	Certified Design Material
6. The control room has its own dedicated ventilation system. This eliminates the possibility of smoke, hot gases, and fire suppressants, originated in areas outside the main control room, to migrate via the ventilation system to the control room.	Certified Design Material
7. Separate ventilation systems for each division eliminates the possibility of smoke, hot gases, and fire suppressants migrating from one division to another.	Certified Design Material
8. There are no sources of "unlimited" external flooding in the reactor building. The interface between the CCWS and the ultimate heat sink (through the service water system) is located in a separate structure outside the reactor building. <insert 19.15-8A here	Certified Design Material
9. Consequential flooding of safety related plant structures from Turbine Building sources is prevented by the following design features: (a) plant grade below openings to safety related structures; (b) openings to safety related structures above the maximum flood level for the Turbine Building; and (c) site grade such that water would flow away from structures where safety related equipment is located.	CESSAR-DC Sections 3.4 & 10.4.1.3

Additions

Insert 19.15-1A(new bullet in item 1)	
INSIGHT	**DISPOSITION**
Details of the seismic walkdown will be developed by the COL applicant	COL Item ~~19.7.5.3-1~~ 19-8 (CESSAR-DC section 19.7.5.3)

Insert 19.15-2B (new bullet in item 2)	
INSIGHT	**DISPOSITION**
The COL Applicant should compare the as-built SSC HCLPFs to those assumed in the System 80+ seismic margins analysis (SMA). Deviations from the HCLPF values for assumptions in the SMA should be evaluated by the COL Applicant to determine if any vulnerabilities have been introduced	COL Item ~~19.7.5.3-2~~ 19-9 (CESSAR-DC section 19.7.5.3)

Insert 19.15-4A (new bullet for item 4)	
INSIGHT	**DISPOSITION**
The COL applicant should incorporate the list of risk important systems, structures and components (SSCs) as presented in table 19.15.6-1 in its D-RAP and O-RAP.	COL Item ~~19.15.6-1~~ 19-14 (CESSAR-DC section 19.15.6)

Insert 19.15-8A (new bullet for item 8)	
INSIGHT	**DISPOSITION**
The seals for the underground pipe chase (contains CCW piping) between the nuclear annex and the CCW building will be capable of withstanding and internal flood from a pipe break in the CCWS/SSWS building (e.g., service water).	Certified Design Material

TABLE 19.15-1

(Sheet 2 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
10. Electrical separation between the two safety-related divisions is maintained.	Addressed in all safety-related systems
11. 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. The drains are sized to handle the potential discharge of fixed fire suppression systems and fire hoses.	Certified Design Material CESSAR-DC Sections 3.4 & 9.3.3
12. 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 (Insert 19.15-12A here)	COL ITEM ~~[illegible]~~ (CESSAR-DC Section 19.8.1.2) 19-18
13. The grid system for System 80+ will include at least two preferred power circuits, each having sufficient capacity. They will be continuously energized and available to provide power to safety related loads. The two designated offsite power transmission lines shall be designed and routed to minimize, to the extent practicable, the likelihood of their simultaneous failure. These circuits shall be routed to ensure no single event, such as a tower falling or a line breaking can simultaneously affect both circuits in a way such that neither can be returned to service. The two offsite power circuits shall terminate at two switchyards that are physically separate and electrically independent to the extent practicable.	Certified Design Material 19-17
14. During plant shutdown, risk can be minimized by appropriate outage management, administrative controls, procedures, and operator knowledge of plant configuration. This will be an important COL applicant activity.	COL ITEM ~~[illegible]~~ (CESSAR-DC section 19.8.1.2)
15. Divisional separation exists also between redundant charging pumps and their power supplies and redundant trains of instrument air.	CESSAR-DC Figure 1.2-4 Sections 8.3.1.1.2.1 & 9.3.1.2.1

Addition

Insert 19.15-12A	
INSIGHT	**DISPOSITION**
The COL applicant should incorporate in its configuration control program a requirement that, during modes 4, 5, and 6, the water tight flood doors and fire doors will be maintained closed on at least one quadrant within the subsphere (containing either an SCS or CSS pump) to help prevent common-mode failures from internal floods or fires. The SCS or CSS pump in this quadrant shall be operable. If the flood or fire doors to this quadrant must be opened for reasons other than to permit normal access a fire watch will be established for the affected door.	COL Item ~~19.8.1.2.2~~ 19-18 (CESSAR-DC section 19.8.1.2)

TABLE 19.15-1

(Sheet 3 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
16. The COL applicant will develop procedures for manually aligning the alternate AC power supply (gas turbine) when one of the two diesel generators is unavailable during a loss of offsite power event.	COL ITEM 19-19
Breakers between the Permanent Non-Safety (PNS) and the class 1E buses will be interlocked so that a PNS bus cannot be aligned to a class 1E bus that is being powered by an EDG.	Certified Design Material
17. 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 provides the Manual Hardwired ESFAS Actuation System for the controls and for display there are Hardwired Key Indications of Critical Function Status for post accident monitoring.	Certified Design Material

TABLE 19.15-1

(Sheet 8 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
22. The following are some important aspects of the Electrical Distribution System (EDS) as represented in the PRA:	
The EDS includes features intended to reduce the frequency of loss of offsite power (LOOP) events and station blackout (SBO) events.	
Replace with Insert 19.15-22A ~~The turbine generator system and the associated buses are designed to run back to maintain "hotel" load on a loss of load.~~	~~CESSAR-DC Section 7.7.1.1.6~~
The two emergency diesel generators are provided with dedicated 125V DC batteries (DC Division Batteries). Therefore they can start and load without the emergency channel batteries.	CESSAR-DC Section 8.3
In addition to the two emergency DGs, the System 80+ design has an alternate standby onsite AC power source. This is a non-safety combustion turbine power source which is independent and diverse from the DGs.	Certified Design Material
The two EDGs are physically and electrically isolated from each other.	Certified Design Material
Each of the six independent load group channels and divisions of 125 V DC Vital Instrumentation and Control Power is provided with a separate and independent class 1-E 125 V battery (2 Division Batteries and 4 channel Batteries). Each battery is sized to supply the continuous emergency load of each own load group for a period of 2 hours.	Certified Design Material
The six independent and separate class 1-E 125 V DC batteries permit operating the I & C loads associated with the turbine-driven emergency feedwater (EFW pumps for 8 hours, assuming manual load shedding or the use of a load management program. This enhances the Station Blackout (SBO) coping capability of the System 80+ design.	CESSAR-DC Section 8.3
Each Emergency Diesel Generator (EDG) has a complete and separate fuel oil storage system. The storage system has sufficient fuel to permit EDG operation for no less than 7 days.	Certified Design Material
Each EDG has two independent air starting systems.	Certified Design

Addition

Insert 19.15-22A	
INSIGHT	**DISPOSITION**
The turbine generator system and its associated buses are designed to run back to maintain 'hotel' load on a loss of load.	CESSAR-DC section 7.7.1.1.6
The run back feature of the turbine generator system will be included in the D-RAP.	D-RAP

TABLE 19.15-1

(Sheet 9 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
23. The following are some important aspects of the Station Service Water System (SSWS) and the Component Cooling Water System (CCWS) as represented in the PRA: Replace with Insert 19.15-23A	
~~Each of these systems (i.e., CCWS and SSWS) has two redundant and separate safety related divisions with heat dissipation capacity to achieve and maintain safe shutdown. Each division has two pumps.~~	~~Certified Design Material 2.7.6~~
Typically during normal operation one SSW and one CCW pump in each division are running with the second pump of SSW and CCW in standby. The standby pump will automatically start if the running pump in that division trips. This configuration reduces the demand failures of pumps and valves which were found to be significant contributors to risk in current generation plants with standby CCWS/SSWS designs.	CESSAR-DC Sections 9.2.1.2.2, 9.2.1.2.2.2 & 9.2.2.2.1.2
The supply and return lines in one division of the SSWS are completely separated from the supply and return lines of the redundant division.	Certified Design Material
SSWS valves in the supply and return lines are locked in the desired position so that only actuation of the pumps are required to place a division in service.	CESSAR-DC Figure 9.2.1-1, Sheets 1 & 3
The ESF actuation System signals isolate the non-safety related portion of the CCWS following an accident condition, except for cooling for the RCPs, IAS compressor coolers, charging pump motor coolers, and charging pump miniflow heat exchangers.	Certified Design Material

Addition

Insert 19.15-23A	
INSIGHT	**DISPOSITION**
Each of these systems (i.e., CCWS and SSWS) has two redundant and separate safety related divisions with heat dissipation capacity to achieve and maintain safe shutdown. Each division has two pumps. The two CCW Heat Exchanger Buildings (one per division) and the SSW Structure are seismic category 1 structures (and the divisional walls of the SSW structure are part of the structure).	Certified Design Material
The Station Service Water Pump structure will be designed such that an internal fire or internal flood on one side of the divisional wall will not affect the other division (e.g., by propagation or by causing failure of the divisional wall	Certified Design Material Interface

TABLE 19.15-1

(Sheet 11 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
26. Aggressive Secondary Cooldown (ASC), which involves cooling of the RCS by opening the ADVs and ensuring that EFW is being delivered to both steam generators given failure of safety injection, has a significant impact on the core damage frequency contribution for small LOCAs and SGTR. Given a small LOCA or SGTR with failure of Safety Injection, the SCS can be aligned to provide the injection function if the RCS is depressurized to the SCS pump shut off head.	Certified Design Material
ABB-CE will provide EPG guidance for the use of the EFWS, and the TBS or ADVs for ASC and the alignment of the SCS for injection operation.	EOGs
27. The following are features of the System 80+ control room design which were assumed to minimize risk from fires in the control room:	
The materials in the control room panels ~~are fire retardant.~~ do not independently support combustion	CESSAR-DC Section 7.7.1.3.1
The energy sources coming into the control panels are limited to low power voltage to the maximum extent practical, thus practically eliminating potential ignition sources within the panels.	CESSAR-DC Section 7.7.1.3.1
A significant portion of the control and indication signals are interfaced to the main control panel via fiber optic cables.	Certified Design Material
28. Sufficient instrumentation and controls are provided at the Remote Shutdown Panel to bring the plant to safe shutdown in case the main control room must be evacuated. Indication and control are provided for EFW, SCS, ADVs, SIS, RDS, CCWS, and SSWS ITAAC.	Certified Design Material
Equipment that do not have dedicated instrumentation and controls at the Remote Shutdown Panel can be controlled via the operator's module. This provides the ability to control most plant functions, albeit on a limited basis, from the Remote Shutdown Panel.	CESSAR-DC Section 7.4

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
29. A control room fire will not impact the instrumentation and controls located at the Remote Shutdown Panel, or the equipment which is required to place the plant in cold shutdown, due to the following features of the System 80+ design:	
The main control room and the remote shutdown room are located at different elevations and in different fire areas.	Certified Design Material
The main control room ventilation system is different from the ventilation system for the remote shutdown room.	Certified Design Material
The stairwells connecting the main control room and the remote shutdown room are pressurized, thus not allowing smoke, hot gases and fire suppressants to migrate from one room to the other.	CESSAR-DC Section 9.4, Figures 1.2-5A through 1.2-9
The main control room is continuously pressurized to prevent the entry of smoke, hot gases, dirt and fire suppressants from other areas.	Certified Design Material
30. All fire barriers which provide separation between the two divisions are rated for at least 3 hours.	Certified Design Material
It was assumed that all fire doors and penetrations within the fire barriers are maintained with high reliability during power operation to prevent the propagation of fire from one area to the next.	D-RAP
31. The possible sources of internal flooding within the Nuclear Annex and Reactor Building are located below elevation 70+0.	Certified Design Material
32. Solid state switching devices and electro-mechanical relays resistant to relay chatter will be used in the Nuplex 80+ protection and control systems. Use of these devices and relays either eliminates or minimizes the mechanical discontinuities associates with similar devices at operating reactors.	CESSAR-DC Sections 7.1.1.7, 7.2.1.1 & 7.3.1.1 COL ITEM 8 (Relay Chatter Resistance)
33. The Startup Feedwater System (SFWS), a non-safety related system, can be used to deliver feedwater to the SGs following a reactor trip. The SFWS pump is powered from the Permanent Non-Safety (PNS) bus and can be powered by the AAC. The SFWS pump can be aligned to the CST or the deaerator storage tank. With alignment to either storage facility, the NPSH for the pump is adequate to prevent pump cavitation and failure.	CESSAR-DC Sections 10.4.7.2.3 & 10.4.7.2.4 Figure 8.3.1-1

19-20

TABLE 19.15-1

(Sheet 13 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
34. There will be a diverse RCP seal injection capability using a positive displacement pump that is diverse from the CVCS and can be powered from either the EDG or the AAC. <Insert 19.15-34A here>	Certified Design Material
Replace with Insert 19.15-35A 35. The functions of the Emergency Containment Spray Backup System are to provide an independent self-contained means of supplying water to the containment spray header for heat removal from the containment atmosphere during emergency conditions when the Containment Spray System and the Shutdown Cooling System pumps are not available and to provide scrubbing of radioactive materials from the containment atmosphere.	CESSAR DC 19.6.3.15
The CS headers can accept spray flow from an external source of water via an 8 inch "T" connection.	CESSAR DC Figures 6.3.2-1A & 1B
The accident management procedures will address use of the Emergency Containment Spray Backup System.	COL ITEM
36. There is a Hydrogen Mitigation System (HMS) utilizing ignitors to control hydrogen during a severe accident.	Certified Design Material
The accident management procedures will address use of the HMS.	COL ITEM 19-16 (CESSAR-DC section 19.15.6)
37. The Hydrogen purge Vent to the annulus is not credited in the PRA. However, the use of this vent could decrease the late containment failure probability.	EOGs
38. Each half of the subsphere is compartmentalized to separate redundant safe shutdown components, to the extent practicable while maintaining accessibility requirements. The subsphere, which houses the front line safety systems is compartmentalized into quadrants, with two quadrants on either side of the divisional structural wall. Flood barriers provide separation between quadrants, while maintaining equipment removal capability. Emergency feedwater pumps are located in separate compartments within the quadrants with each compartment protected by flood barriers. Flood barriers also provide separation between electrical equipment and fluid mechanical systems at the lowest elevation within the Nuclear Annex.	Certified Design Material
Elevated equipment pads prevent equipment from being inundated in the event of flooding. <Insert 19.15-38A	CESSAR-DC Section 3.4.4.1

Addition

Insert 19.15-34A	
INSIGHT	**DISPOSITION**
The alternative positive displacement seal injection pump is located in such a manner as to minimize its vulnerability to internal floods and fires that could also affect the primary means of providing RCP seal cooling or RCP seal injection.	CESSAR-DC figure 1.2-5A

Insert 19.15-35A		
	INSIGHT	**DISPOSITION**
35.	An emergency containment spray backup function provides a means of supplying water to the containment spray header from a station AC independent external source.	Certified Design Material
	The final design of the ECSBS is not completed. The design of the ECSBS is envisioned to include the following design features: 1) an 8-inch diameter "tee" connection to the containment spray recirculation line (2) an extension of 8-inch diameter Class 2 piping from the "tee" connection from the containment spray recirculation line to the exterior of the Nuclear Annex, (3) external connections for temporary hookup of an external source of water that are located at or near grade, (4) a portable pumping source (e.g., fire truck) that is independent of site AC power buses. This pumping device will be capable of supplying sufficient flow to the containment spray header at 24 hours after a severe accident to provide sufficient heat removal capability via the spray droplets to prevent the containment pressure from exceeding the service level C pressure. Preliminary calculations indicate a flow rate of 750 gpm would be sufficient. and (5)all necessary hoses, fittings and spool pieces would be stored with the pumping device or at or near the "tee" connections.	CESSAR-DC sections 6.5.2.5.1 and 19.11.2.8
	The detailed system design and location of all associated valves and connections should take into account expected radiation levels and shielding requirements for any required local operator actions.	COL Item ~~19.11.3.8-2~~ 19-11 (CESSAR-DC section 19.11.3.8)
	The specific flow rate for the pumping device will be determined as part of the detailed design.	COL Item ~~19.11.3.8-1~~ 19-10 (CESSAR-DC section 19.11.3.8)
	Detailed procedures for use of the system will be developed by the COL applicant	COL Item ~~19.15.6-3~~ 19-16 (CESSAR-DC 19.15.6)

Addition

Insert 19.15.38A	
INSIGHT	**DISPOSITION**
There are three-hour fire barriers as well as flood barriers between quadrants in the subsphere.	Certified Design Material
Within each division, there are two Class 1E 4160 KV switchgear. These are separated by three hour fire barriers and are arranged to be associated with one of the subsphere quadrants. Power cables from the diesel generator room in a given division to their associated switchgear are fully separated and the cables from the switchgear to their associated pumps are fully separated.	CESSAR-DC 9.5.1.14

TABLE 19.15-1

(Sheet 14 of 14)

SIGNIFICANT PRA-BASED SAFETY INSIGHTS FOR SYSTEM 80+

INSIGHT	DISPOSITION
39. Flood protection is integrated into the floor drainage systems. The floor drainage systems are separated by division and Safety Class 3, Seismic Category 1 valves which prevent backflow of water to areas containing safety related equipment. Each subsphere quadrant contains its own separate sump equipped with redundant Safety Class 3, Seismic Category 1 sump pumps and associated instrumentation. These pumps are also powered from the diesel generators in the event of loss of offsite power. The Nuclear Annex also has its own divisionally separated floor drainage system, having no common drain lines between divisions.	Certified Design Material
Floors are gently sloped to allow good drainage to the divisional sumps. Floor drains are routed to the lowest elevation to prevent flooding of the upper elevations. The lowest elevation in each division has adequate volume to collect water from a break in any system without flooding the other division. In addition, potential discharge of fixed fire suppression systems and fire hoses is considered in the sizing of floor drains to preclude flooding of areas should the fire protection systems be initiated.	CESSAR-DC Section 9.3.3
40. The COL ~~will~~ should maintain a well trained and prepared fire brigade.	COL ITEM ~~[illegible]~~ 19-13 (CESSAR-DC Section 19.15.2)
41. The System 80+ low pressure systems which interface with the RCS are protected against ISLOCA by a combination of increases in the piping pressure limits and autoisolation capability based on pressure sensors.	Certified Design Material

<INSERT 19.15-1 NEW here)

Addition

Insert 19.15-1NEW Additional Items for inclusion in Table 19.15-1	
INSIGHT	**DISPOSITION**
42. The COL applicant should consider the information on risk important operator actions from the PRA, as presented in Table 19.15.6-2, in developing and implementing procedures, training and other human reliability related programs.	COL ITEM ~~19.15.6-2~~ ← 19-15 (CESSAR-DC 19.15.6)
43. During detailed design phase, the COL applicant should update the PRA using the final design information and site specific information. As deemed necessary, the COL applicant should update the PRA, including the shutdown risk evaluation, and the internal fire and flood evaluation. Based on site specific information, the COL applicant should also re-evaluate the qualitative screening of external events. If any site specific susceptibilities are found, the applicable external event should be included in the updated PRA.	COL Item ~~19.15.1.1~~ ← 19-12 (CESSAR-DC section 19.15.1)
44. The structure that houses the combustion gas turbine must have a HCLPF of at least that of the gas turbine itself, or must be designed in such a manner so that failure of this structure following a seismic event up to HCLPF of the gas turbine will not affect the operability of the gas turbine .	CESSAR-DC 19.7.5.3
45. During the HFE V&V, the risk significance of tasks impacted by findings will be considered in the finding resolution process. The resolution process will qualitatively confirm that the findings, as dispositioned, will not lead to a risk-significant increase in error potential from that represented in the HRA, or additional risk-significant errors not modeled in the HRA.	Human Factors Engineering Verification and Validation Plan for NUPLEX 80+, NPX80-IC-VP790-03, Section 8.1 (supports Certified Design Material)
46. No water lines are routed above or through the control room and the computer room. HVAC water lines contained in rooms around the control room are located in rooms with raised curbs to prevent leakage from entering the control room.	CESSAR-DC 3.4

Addition

47.	A reactor cavity flood system is provided to enhance the coolability of ex-vessel core debris Procedures for use of the cavity flood system during a severe accident will be developed by the COL applicant as part of their plant-specific severe accident management procedures. The reliability of the cavity flood system and associated valves is important. The COL applicant will ensure the reliability of the cavity flood system	Certified Design Material COL Item 19-15 ~~19.15.6-2~~ (CESSAR-DC 19.15.6) D-RAP, Table 19.15.6-1
48.	Containment integrity is important to reduce the risk to the public. The major containment penetrations (equipment hatch, personnel, airlocks and fuel transfer tube) will be designed to assure that they will not fail up to ASME service level "C" for the containment shell. 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. Containment integrity is important to reduce the risk to the public. The major containment penetrations (equipment hatch, personnel, airlocks and fuel transfer tube) will be designed to assure that they will not fail up to ASME service level "C" for the containment shell. 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.	D-RAP, Table 19.15.6-1 D-RAP, Table 19.15.6-1
49.	The reliability of the MSSVs, ADVs, and MSIVs is important. The COL-applicant will ensure the reliability of these components	D-RAP, Table 19.15.6-1
50.	Flood barriers separating the flood zones in the nuclear annex, the CCWS Heat Exchanger buildings and the SSWS pump structure are designed to withstand water pressure generated by internal flooding . Flood barriers, including water tight doors and penertations, will be addressed in the O-RAP	Certified Design Material D-RAP, Table 19.15.6-1

19.15.3 EXTERNAL EVENTS RISK PROFILE INSIGHTS

The external events analyses for the System 80+ design included both qualitative and quantitative analyses. Bounding site characteristics were used for the quantitative analyses to minimize potential future restrictions on plant siting. The qualitative external events evaluation involved the following: (1) identification of the external events to be considered, (2) grouping of events with similar plant effects and consequences, (3) establishment of screening criteria to eliminate events that are insignificant contributors to risk, and (4) identification of events that require further quantitative evaluation. Based on the qualitative evaluation, most of the external events were eliminated from further quantitative evaluation. Four external events (tornado, fire, flood, and seismic) were identified as having the potential to induce system failures and therefore required further quantitative evaluation.

The major findings and insights obtained for tornados, fires, floods, and seismic events are provided in Sections 19.15.3.1, 19.15.3.2, 19.15.3.3, and 19.15.3.4 respectively.

19.15.3.1 Insights from the Tornado Strike Analysis

The core damage frequency due to tornado strike events is calculated to be 2.5E-07 per year. The dominant contributors to the core damage frequency of tornado strike events are provided in Table 19.15.3-1. For the System 80+ PRA, the following assumptions were made for the tornado strike accident sequences.

A. It was assumed that offsite power will be lost for more than 24 hours and the plant will rely on the emergency diesel generators during this period.

B. The turbine/generator will be unable to run back and pickup hotel loads following a tornado strike.

C. The alternate AC power source was conservatively assumed to be unavailable following a tornado strike.

D. The Station Service Water System (SSWS) intake structure was assumed to be vulnerable to accumulation of debris due a tornado strike event and was included in the models. The protection of the SSWS intake structure against the accumulation of debris due to a tornado strike could prevent or minimize the loss of suction to the SSWS pumps.

E. Safety related structures outside the nuclear island will not be destroyed by a tornado strike.

Insert 19.15.3.1A here

Insert 19.15.3.1A

The COL applicant should re-evaluate the vulnerability of the SSWP intake to tornado-generated debris. [COL Item 1~~9.1.2.2.2-1~~]
19-1

N. The reliabilities of the fire detection and suppression systems are assumed to be at least 80% and 96%, respectively.

< Insert 19.15.3.2A here >

A quantitative assessment of the risk due to internal fires can not be made at this time because detailed design information for cable routing and the fire detection and fire suppression system is not presently available. However, a scoping evaluation is performed to assess the risk due to internal fires in areas of the Nuclear Annex other than the containment or the control room. Two types of fires were considered in the scoping evaluation: (1) a fire in an area which could disable safety-related equipment in that area and which has the potential for initiating a transient, and (2) a fire in an area which by itself could disable safety-related equipment but would require the penetration of a fire barrier in order to initiate a transient. The first type of fire is designated as type "a" and the second type as type "b". The fire ignition sources and frequencies by applicable areas are presented in Table 19.15.3-2.

Although a detailed quantitative analysis of internal fires was not performed at this stage of the System 80+ design, a scoping estimate of the risk due to fire was calculated by using a conservative scoping value (4.6E-02 per year) for fire event frequency and by assuming that the effects on plant systems would be the same as a loss of one division of component cooling water/station service water. Using this approach, the estimated scoping value core damage frequency due to internal fires is 6.1E-08 per year and the sequence of importance involves an internal fire followed by failure of long-term decay heat removal and failure of SDS.

Based on the robust seal design for the RCPs used in the System 80+ design and on the results of tests and operating experience, ABB-CE asserts that the RCP seals will not fail on loss of seal injection and seal cooling. However, in the interests of completeness, an assessment of a postulated fire induced RCP seal LOCA was included as part of the quantitative fire scoping evaluation. The scoping value for core damage frequency associated with the postulated fire induced seal LOCA was calculated to be 5.2E-10 per year. The potential risk due to a postulated fire inside containment was also assessed. The estimated scoping value of core damage frequency due to fire inside containment is 1.3E-09 per year. Thus, the total estimated scoping value of core damage frequency for internal fires is 6.3E-08 per year.

The following insights were drawn from the internal fire scoping assessment:

Insert 19.15.3.2A

O. Althought fire brigade action to suppress fires was not modeled in the scoping fire risk evaluation, the capabilities of the plant fire brigade are important to maintaining a low fire risk. The COL applicant should maintain a a well trained and prepared fire brigade. [COL Item ~~19.15.3.2-1~~]

19-13

New Section 19.15.6
old Section 19.15.6 changed to 19.15.7

19.15.6 RISK SIGNIFICANT SSCs FOR CONSIDERATION IN THE D-RAP

Table 19.15.6-1 presents a list of risk significant Systems Structures and Components (SSCs) that should be included in the D-RAP as described in section 17.3. The COL applicant should incorporate these SSCs in their O-RAP.[COL Item 19-14] These SSCs were selected based on their risk importance as determined in the level 1 analyses, the level 2 analyses, the level 3 analyses, the shutdown risk evaluation, the internal fire and flood evaluation, and the seismic margins evaluation. For the level 1 analyses and the shutdown risk analyses, systems and components were included as risk significant if their Risk Achievement Worth (RAW) was greater than or equal to 5.0 or their Risk Reduction Worth (RRW) was 1.10 or greater. SSCs with an RAW between 2.0 and 5.0 were selected if their RRS was greater than 1.05. For the Seismic Margins Assessment, an SSC was included if it was a dominant contributor to the Plant HCLPF. For the level 2, level 3, and internal fire and flood analyses, items were included based on engineering judgement. SSCs were also included in the list if specific engineering commitments were made by the system designers. Table 19.15.6-1 contains three columns. The first column identifies the system, structure or component. The second column presents the rationale (basis) for including the SSC in the D-RAP (i.e., RAW > 5.0, level 2 considerations, engineering judgement, engineering commitment, etc.) The third column briefly describes the item and any associated insights. The third column also identifies any test interval or maintenance assumptions that were used in the PRA. This table does not include any failure rate or unavailability information. All component failure rates are documented in section 19.5 and its associated appendices. The random failure rates for the individual components for the specific failure modes of concern are summarized in table 19.5-2. The common cause failure rates are summarized in table 19.5-3. The maintenance unavailabilities are summarized in table 19.5-4. The component and structure HCLPF values are summarized in tables 19.7.5.1-1 and 19.7.5.1-2. Table 3.9-15 summarizes the In-Service Testing program for all safety related pumps and valves and presents the applicable test intervals.

Table 19.15.6-2 presents a list of Important Operator Actions selected from the PRA. These operator actions were selected based on their risk importance as determined in the level 1 analyses, the level 2 analyses, the level 3 analyses, the shutdown risk evaluation, the internal fire and flood evaluation, and the seismic margins evaluation. For the level 1 analyses and the shutdown risk analyses, operator actions were included as important if their Risk Achievement Worth (RAW) was greater than or equal to 5.0 or their Risk Reduction Worth (RRW) was 1.10 or greater. Operator actions with an RAW between 2.0 and 5.0 were selected if their RRW was greater than 1.05. For the Seismic Margins Assessment, an operator action was included if failure to perform that action could result in a lower overall plant HCLPF value. For level 2, level 3, and internal fire and flood analyses, items were included based on engineering judgement.

The COL applicant is responsible for developing all plant procedures. These procedures include, but are not limited to, the normal operating procedures, system operating procedures, maintenance procedures, emergency operating procedures and severe accident procedures. The Emergency Operating Guidelines (EOGs) provide provide guidance to the COL applicant for developing the detailed Emergency Operating Procedures. Appendices to the EOGs provide guidance on severe accident procedures and emergency operating considerations during shutdown operations. In developing and implementing procedures, training and other human reliability related programs, the COL applicant should consider the information on risk important operator actions presented in table 19.15.6-2. [COL Item 19-15]

In the severe accident management procedures, the COL applicant should include procedures for the use of the Cavity Flood System, the Hydrogen Mitigation System, and the Emergency Containment Spray Backup function of the CSS. [COL Item 19-16]

The COL should develop procedures for manually aligning the Alternate AC power supply when one of the two emergency diesel generators is unavailable during a loss of offsite power. [COL Item 19-19]

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Emergency Feedwater System (EFWS)	RAW* > 5.0	The EFWS is used for secondary side heat removal following a transient or small LOCA. The EFWS consists of two trains, one per steam generator (SG). Each train has two 100% capacity redundant and diverse pumps, one motor driven and one turbine driven. Each train has its own EFW storage tank which can be refilled from the condensate storage tank or the demineralized water makeup system. There is a cross connect between the two trains on the discharge side of the pumps. The cross-connect line is isolated by two NC manual isolation valves. In the PRA, only unscheduled maintenance was assumed. Maintenance unavailability was assigned at the subtrain level based on generic data provided in the EPRI PRA Key Assumptions and Groundrules (KAG) document[7].
Component: EFW Motor driven pumps	RAW > 5.0 (CCF)	The EFW pumps deliver EFW to the SGs for secondary side decay heat removal.. Each train has one 100% capacity motor-driven pump powered from the appropriate vital 4.16 KV bus. Consistent with current practices, the EFW pumps are tested on a quarterly basis with unscheduled maintenance performed on failure. Maintenance unavailability is covered by the subtrain level maintenance unavailability (see above).

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> EFW Turbine driven pumps	RAW > 5.0 (CCF)	The EFW pumps deliver EFW to the SGs for secondary side decay heat removal.. Each train has one 100% capacity turbine-driven pump powered by steam derived from its associated SG. Consistent with current practices, the EFW pumps are tested on a quarterly basis with unscheduled maintenance performed on failure. Maintenance unavailability is covered by the subtrain level maintenance unavailability (see above).
<u>Component:</u> EFW pump discharge check valves	RAW > 5.0 (CCF)	Consistent with current practices, these check valves are assumed to be tested on a cold shutdown basis with maintenance performed on failure.
<u>Component:</u> EFW distribution line check valves	RAW > 5.0 (CCF)	These check valves are assumed to be tested on a cold shutdown basis with maintenance performed on failure.
<u>Component:</u> EFW distribution line AC motor operated valves	RAW > 5.0 (RAW)	These valves must open to deliver EFW to the respective SG in the event of a transient. These valves are also used to control EFW flow to the SGs during long term EFW usage. Consistent with current practices, these valves are tested on a quarterly basis with unscheduled maintenance performed on failure. The maintenance unavailability is covered by the subtrain maintenance unavailability.
<u>Component:</u> EFW Storage Tank (EFWST)	SMA, Flood	The EFWSTs provide the inventory for the EFW system. Seismic failure of the wall between the EFWST and the adjacent DG room would result in flooding of the DG room with failure of the DG and the loss of EFW inventory. Each EFWST also represents a potential flood source for the subsphere for its associated division.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>**Component:**</u> Condensate Storage Tank (CST)	SMA	If the EFW system is used for extended time periods, the CST provides a source to replenish the EFWST inventory. Seismic failure of this tank would preclude extended EFW usage following a seismic event . The seismic fragility (CDFM HCLPF) used for this tank was 0.56g.
<u>**Component:**</u> EFW storage tank inlet valves	RAW > 5.0 (CCF)	Generic demand failure rate used. No specific assumptions on test or maintenance intervals
<u>**Component:**</u> EFW motor driven pump breakers	RAW > 5.0 (CCF)	The motor -driven EFW pump circuit breakers must close to provide power to the motor-driven EFW pumps. The breakers are tested at the same time the motor-driven pumps are tested.
<u>**Component:**</u> EFWST fill line manual isolation valve between CST and EFWSTs	RAW > 5.0	Generic demand failure rate used. No specific assumptions on test or maintenance intervals
<u>**Component:**</u> EFWST fill line check valve between CST and EFWSTs	RAW > 5.0	Generic demand failure rate used. No specific assumptions on test or maintenance intervals

TABLE 19.15.6-1
RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP

SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Electrical Distribution System (EDS)	RAW > 5.0	The EDS is provided to supply AC and DC electrical power necessary for normal plant operation and mitigation of abnormal events. The EDS consists of two portions, the non-class 1E portion which provides power to equipment needed for normal operation and equipment not needed for safe shutdown, and the class 1E portion, consisting of two class 1E divisions which provides power to equipment needed to establish and maintain safe shutdown. During normal operation, station power is provided from the grid via one of two offsite power circuits with automatic transfer to the second source on the permanent non-safety buses if the first source is lost. There are manual transfer capabilities to power the 1E buses directly from the reserve auxiliary transformers. If the grid is lost, the turbine generator can runback and pick up hotel load. If this is unsuccessful, AC power to the class 1E loads can be supplied by the two emergency diesel generators (1 per class 1E division). Selected non-class 1E loads on the permanent non-safety bus can be powered from the standby combustion turbine. The standby combustion turbine is capable of supplying all the loads on the permanent non-safety bus plus all the safety loads on one of the two class 1E buses.
Component: 125 VDC class 1E vital buses	RAW > 5.0 (CCF)	The class 1E 125 VDC buses provide safety grade control and instrumentation power. These buses are continuously energized and faults are detected on occurrence. Unscheduled maintenance is performed on failure.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> 480 VAC class 1E load center transformers	RAW > 5.0 (CCF)	The class 1E 480 VAC load centers provide 480 VAC power to the 480 VAC Motor Control Centers (MCCs) for safety related 480 VAC loads. The load centers are supplied with power from the 4.16 KV buses via the load center transformers. The load center transformers are continuously energized and faults are detected on occurrence. Unscheduled maintenance is performed on failure.
<u>Component:</u> 480 VAC class 1E load centers	RAW > 5.0 (CCF)	The class 1E 480 VAC load centers provide 480 VAC power to the 480 VAC Motor Control Centers (MCCs) for safety related 480 VAC loads. The load centers are continuously energized and faults are detected on occurrence. Unscheduled maintenance is performed on failure.
<u>Component:</u> 480 VAC class 1E Motor Control Centers	RAW > 5.0 (CCF)	The 480 VAC class 1E MCCs provide power to the various 480 vac safety loads. The 480 VAC MCCs are normally energized with the breaker(s) or contactors to the safety load(s) open to remove power from the load. The load would be energized by closing the breaker(s) or contactors. Faults in the MCCs are detected on occurrence and unscheduled maintenance is performed on failure.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> 4.16KV class 1E Buses	RAW > 5.0 (CCF)	The 4.16KV buses provide the AC power to AC-powered safety related loads. 4.16KV power for the pump motors is provided directly from the 4.16 KV buses. 480 VAC power is provided to the load centers via load center transformers. The 4.16 KV buses also indirectly provide power to the vital DC buses via the battery chargers which are powered from 480 VAC vital MCCs. The 4.16 KV buses are continuously energized and faults are detected on occurrence. Unscheduled maintenance is performed on failure.
<u>Component:</u> 4.16KV Permanent Non-Safety buses	RAW > 5.0 (CCF)	The 4.16 KV Permanent Non-Safety (PNS) bus provides 4..KV power and 480 VAC power via stepdown transformers to the permanent non-safety loads. The 4.16 KV PNS bus is continuously energized and faults are detected on occurrence. Unscheduled maintenance is performed on failure.
<u>Component:</u> 125 VDC class 1E vital batteries	RAW > 5.0 (CCF)	The 125 VDC batteries provide 125 VDC power to the 125 VDC vital buses in the event that AC power is unavailable. During normal operation, the battery chargers maintain a floating charge on the batteries. Consistent with current standard practices, the battery terminal voltage is assumed to be verified every 7 days.

TABLE 19.15.6-1
RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP

SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>**Component:**</u> Emergency Diesel Generators (EDGs)	RAW > 5.0 (CCF)	The EDGs supply 4.16 KV power to the Class 1E loads in the event that offsite power is not available following a transient or accident. The EDGs are tested on a monthly basis. Unscheduled maintenance is performed on failure. Maintenance unavailability was calculated based on the EDG failure rate, a monthly test interval, and an allowed outage time of 72 hours.
<u>**Component:**</u> Emergency Diesel Generator Load Sequencers	RAW > 5.0 (CCF)	The Engineered Safety Features Component Control System has the load sequencers for the vital 4.16 KV buses. They protect the DG from overload and also prevent vital buses from all loading at once if offsite power is lost and regained. The load sequencers are implemented in the Programmable Logic Controllers (PLCs). The PLCs have internal diagnostic tests on a continuous basis and failures are annunciated. Maintenance is performed on failure.
<u>**Component:**</u> Emergency Diesel Generator Supply Breakers to 4.16KV class 1E buses	RAW > 5.0 (CCF)	These breakers connect the EDGs to the 4.16 KV vital buses. These breakers are tested in conjunction with the monthly EDG tests.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: Alternate AC source (Combustion Turbine)	Engineering Judgement	The Alternate AC (AAC) source will supply 4.16 KV power to the permanent non-safety bus in the event that offsite power is unavailable. In addition, the AAC source can also supply power to one division of class 1E loads if the EDGs are unavailable. The PRA used a generic unavailability for the AAC source as provided in the EPRI PRA Key Assumptions and Groundrules[7]. No specific assumptions were made as to testing and maintenance intervals for the AAC.
Component: DG room ventilation fans	RAW > 5.0 (CCF)	The DG room ventilation system is temperature actuated. Based on operating experience information, it is anticipated that the DG room ventilation system will be actuated when the DGs are started for their monthly testing. Thus, the DG room ventilation fans are assumed to be effectively tested on a monthly basis in conjunction with the DG test.
Component: DG room ventilation dampers	RAW > 5.0	The DG room ventilation system is temperature actuated. Based on operating experience information, it is anticipated that the DG room ventilation system will be actuated when the DGs are started for their monthly testing. Thus, the DG room ventilation dampers are assumed to be effectively tested on a monthly basis in conjunction with the DG test.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> EDG air starting system	Engineering Judgement	Each EDG is assumed to have two independent air starting systems. The starting capability of these systems are tested in conjunction with the monthly EDG test. The replenishment capability of the starting air system is constantly verifiable because the starting air compressors must function to supply air to the starting air tanks due to leakoff and EDG panel usage.
<u>Component:</u> EDG fuel oil storage systems	Engineering Judgement	Each EDG has a complete and separate fuel oil system. These fuel oil systems were assumed to be inspected, tested and maintained consistent with current practices.
<u>Component:</u> Turbine generator	Engineering Judgement	The turbine generator system is designed to be capable of running back to and maintaining "hotel" load following a loss of load.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Component Cooling Water(CCWS) /Station Service Water (SSWS) Systems	RAW > 5.0	The CCWS is a closed loop system that provides cooling water flow to remove heat released from plant systems and components. The CCWS also provides cooling water flow for decay heat removal from the SCS during shutdown cooling and from the CSS flow during containment spray operation. The CCWs rejects the heat to the SSWS via the SSWS heat exchangers. The SSWS is an open loop system, which takes suction from the ultimate heat sink, passes the flow through SSWS heat exchanger to remove the heat from the CCWS, and then discharges the heated water to the ultimate heat sink. The CCWS and SSWS each consist of two divisions. Each division of SSWS and CCWS have two 100% capacity pumps. One pump in each division is normally operating and the other pump is in standby. If the operating pump trips, the standby pump would be started. Each division also has two 100 % capacity SSWS/CCWS heat exchangers with one in service and the other in standby. Manual valve alignment is required to valve in the standby heat exchanger. Pump and heat exchanger maintenance is performed on the pump or heat exchanger that is in standby. The SSWS and CCWS are in operation during normal power operation and faults in operating equipment are detected on occurrence.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> CCWS inlet flow control valves to SCS heat exchangers	RAW > 5.0 (CCF)	These valves open to provide CCWS flow through the shutdown cooling heat exchanger during shutdown cooling operation. It was assumed that, consistent with current practices, these valves were tested on a quarterly basis and unscheduled maintenance performed on failure.
<u>Component:</u> Service Water Intake Structure	Tornado Strike Evaluation	The SSWS uses a common intake structure for both divisions. The intake structure is that structure in which the SSWS draws suction from the ultimate heat sink. The SSWS intake structures are required to meet Reg Guide 1.27 requirements. In the event of a tornado strike on site, the tornado might deposit sufficient debris in the intake structure to cause blockage of the intake structure. If complete blockage occurs, all SSW and CCW flow will be lost. Provisions should be incorporated to protect the intake structure against the accumulation of sufficient debris to block the structure.
<u>Component:</u> CCWS surge tanks	SMA	The CCWS surge tanks are located in the upper levels of the nuclear annex at elevation 170+0. Seismic failure of these tanks could lead to loss of all CCW inventory with subsequent failure of CCW.
<u>Component:</u> CCWS heat exchanger building	SMA	The CCWS heat exchanger buildings are separate from the nuclear annex building. Seismically induced differential sliding of these buildings could result in failure of the CCWS piping from the heat exchanger buildings to the nuclear annex with consequential failure of the CCWS.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: SSWS pump building	SMA, Fire, Flood	The SSWS pump building is a Seismic Category 1 structure which houses the SSWS pumps for both divisions. This structure is outside the CESSAR-DC scope. It was assumed that this structure would have divisional separation equivalent to that in the nuclear annex such that the propagation of internal floods or fires from one division to the other is prevented. It was also assumed that this Seismic Category 1 structure has a seismic strength equivalent to the nuclear annex structure.
SYSTEM: Safety Injection (SI) System	RAW > 5.0	The function SI system is to inject borated water into the RCS to provide RCS inventory control in response to a LOCA or and SGTR. The SI system also provides inventory injection for feed and bleed cooling in conjunction with the RDS.
Component: SI pumps	RAW > 5.0 (CCF)	The SI pumps are required to provide RCS inventory control in response to LOCAs, SGTRs, and situations requiring RCS feed and bleed cooling. The SI pumps also provide long term reactivity control via the injection of borated water. Consistent with current practices, the SI pumps were assumed to be tested quarterly Unscheduled maintenance is performed on failure.
Component: SI pump motor breakers	RAW > 5.0 (CCF)	The SI pump motor circuit breakers are normally open and must close to provide power to the SI pump motors. The breakers are tested at the same frequency as the SI pumps.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> SI pump discharge check valves	RAW > 5.0 (CCF)	Consistent with current practices, these check valves are assumed to be tested on a fuel cycle basis with maintenance performed on failure.
<u>Component:</u> Safety Injection Direct Vessel Injection (DVI) line motor operated valves	RAW > 5.0 (CCF)	The SI DVI MOVs must open to provide injection flow to the DVI lines to provide RCS inventory makeup. Consistent with current practices, these valves are assumed to be tested on a quarterly basis with maintenance performed on failure.
<u>Component:</u> DVI line check valves	RAW > 5.0 (CCF)	This includes all the SI check valves in the DVI line inside containment. These valves must open for injection flow to reach the reactor vessel. The test interval was assumed to be one fuel cycle. Maintenance is performed on failure and only when the plant is shutdown
<u>Component:</u> Hot Leg Injection line check valves	RAW > 5.0 (CCF)	These check valves are in the hot leg injection lines. These valves must open to provide hot leg injection to prevent boron crystallization after initial response to large and medium LOCAs. The test interval for these check valves was assumed to be one fuel cycle. Maintenance on these valves is performed on failure and only when the plant is shutdown.
<u>Component:</u> Hot Leg injection line motor-operated isolation valves	RAW > 5.0 (CCF)	The hot leg injection MOVs must open to provide hot leg injection to prevent boron crystallization after initial response to large and medium LOCAs. Consistent with current practices, these valves are assumed to be tested on a quarterly basis with maintenance performed on failure.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> SIT discharge check valves	RAW > 5.0 (CCF)	In the event of a LOCA, these valves must open for the SIT inventory to inject into the RCS. These valves are tested once a fuel cycle during plant refueling. No scheduled maintenance is performed on these valves while the plant is at power.
<u>Component:</u> Safety Injection Tanks (SITs)	RAW > 5.0 (CCF)	The SITs provide a source of inventory for passive injection into the RCS in response to large and medium LOCAs and during aggressive secondary cooldown for SCS injection for small LOCAs. SIT pressure and level are monitored every 12 hours, but the tanks are inspected and tested only on a fuel cycle basis and all maintenance is performed while the plant is shutdown.
<u>SYSTEM:</u> Engineered Safety Features Actuation System (ESFAS)	RAW > 5.0	The ESFAS provides the signals to actuate equipment in the front line safety systems following a transient or accident. The System 80+ ESFAS was assumed to be as reliable as the System 80 ESFAS. The analysis of the System 80 ESFAS assumed the system was tested on a monthly basis with maintenance performed on failure. For System 80+, most of the ESFAS logic is automatically tested continuously with alarms if problems detected. In addition, a full channel functional test is performed every 92 days to verify that the ESFAS is operable, and to confirm that the automatic testing is functioning properly.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: ESFAS relays	SMA	Solid state switching devices and electromechanical relays will be used in the NUPLEX 80+ protection and control systems. Solid state switching devices are immune to mechanical switching discontinuities. Robust electromechanical relays are selected for NUPLEX 80+ applications such that inherent mechanical contact chatter is within the requisite system performance criteria
SYSTEM: Reactor Protection System (RPS)	Engineering Judgement	The RPS provides the signals to trip the reactor following a transient or accident. Failure to trip the reactor results in an ATWS. The System 80+ RPS was assumed to be as reliable as the System 80 RPS. The analysis of the System 80 RPS assumed the system was tested on a monthly basis with maintenance performed on failure. For System 80+, most of the RPS logic is automatically tested continuously with alarms if problems detected. In addition, a full channel functional test is performed every 92 days to verify that the RPS is operable, and to confirm that the automatic testing is functioning properly.
SYSTEM: Control Room	Engineering Judgement, Fire Evaluation	The plant is operated from the main control room. The control room contains sufficient instrumentation displays and controls to allow the the operators to control the plant during normal operating conditions and to respond to transients and accidents

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: Control Panels	Engineering Judgement, Fire Evaluation	The control panels contain the instrumentation displays and equipment controls needed to control the plant during normal operation and during transient or accident conditions. Materials which do not independently support combustion are used in the control panels to minimze the potential for fires in the control panels propagating to affect multiple channels.
Component: Control room ventilation	Engineering Judgement, Fire Evaluation	The control room has its own dedicated ventilation system. This eliminates the possibility of smoke, hot gases, and fire suppressants originating in areas outside the control room migrating to the control room via the ventilation system
SYSTEM: Remote Shutdown Panel	Engineering Judgement, Fire Evaluation	The Remote Shutdown Panel has sufficient instrumentation and controls to bring the plant to safe shutdown if the main control room must be evacuated.
SYSTEM: Shutdown Cooling System (SCS)	RAW > 5.0	The function of the SCS is to cool the RCS from shutdown cooling entry conditions to cold shutdown conditions. In the event of a small LOCA with failure of the SI system, the SCS can be aligned to provide injection if the RCS is depressurized to below the SCS pump shutoff head.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> Pressure Interlocks for SCS suction valves	RAW > 5.0 (CCF)	The SCS suction Motor Operated Valves (MOVs) are interlocked so that they can not be opened if the RCS pressure is greater than the shutdown cooling entry pressure. Common cause failure of these interlocks would prevent the SCS suction MOVs from opening.
<u>Component:</u> SCS suction MOVs	RAW > 5.0 (CCF)	The SCS suction MOVs must open in order to start shutdown cooling. These valves are located inside containment, are interlocked on RCS pressure and are part of the RCS pressure boundary. Therefore, these valves can not be tested at power. Consistent with current practices, these valves are tested on a cold shutdown basis and maintenance is performed only when the plant is shutdown.
<u>Component:</u> SCS suction isolation MOVs	RAW > 5.0 (CCF)	The SCS suction isolation MOVs must open in order to establish shutdown cooling. These valves are interlocked on RCS pressure, like the suction MOVs , but they are outside containment and are not part of the RCS pressure boundary. It was assumed that these valves could be tested on a quarterly basis. However, the IST in table 3.9-15 of CESSAR-DC, amendment T specifies that these valves are tested on a cold shutdown basis. This difference is addressed in table 19.6A-1
<u>Component:</u> SCS discharge check valves	RAW > 5.0 (CCF)	These check valves must open to establish shutdown cooling flow. These check valves are assumed to be tested on a cold shutdown basis consistent with current practices. Maintenance is performed on failure.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: SCS heat exchanger flow control valves.	RAW > 5.0 (CCF)	The SCS heat exchanger flow control valves must open in order for SCS flow to pass through the heat exchangers to reject the core heat to the CCW and SSW systems. These valves were assumed to be tested on a quarterly basis and maintenance performed on an as needed basis.
Component: SCS pumps	Shutdown Risk Analysis	The SCS pumps recirculate the RCS fluid through the SCS heat exchangers to cool the RCS from shutdown cooling entry conditions to cold shutdown conditions and to maintain cold shutdown conditions. The SCS pumps can be backed up by the CSS pumps during shutdown cooling. The SCS pumps can be used to back up the CSS pumps. The SCS pumps can also be aligned to inject to the RCS if the RCS is depressurized to the SCS pump shutoff head. The SCS pumps were assumed to be tested on a quarterly basis consistent with current practices.
Component: SCS/CSS Crossover Valves	Engineering Judgement, Shutdown Risk Assessment	In order to use the SCS pumps to backup the CSS pumps or to use the CSS pumps to backup the SCS pumps, the SCS/CSS suction and discharge crossover valves must be opened. These valves were assumed to be tested on a quarterly based with maintenance performed on an as needed basis.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Rapid Depressurization System (RDS)	RAW > 5.0	The RDS consists of two trains, each containing two MOVs in series, that provides a discharge path from the top of the pressurizer to the IRWST. The primary function of the RDS is to provide a means of depressurizing the RCS in the event of a severe accident with the RCS at high pressure to prevent a High Pressure Melt Ejection (HPME). The RDS also provides the "bleed" capability for feed and bleed (or once through) cooling of the RCS.
Component: Rapid Depressurization Valves (RDVs)	RAW > 5.0 (CCF)	The RDVs must open for feed and bleed cooling, or for depressurization of the RCS during a severe accident. These valves are inside containment and are part of the RCS pressure boundary so they can not be tested at power. These valves are tested on a fuel cycle basis and maintenance is performed on these valves only when the plant is shut down and depressurized.
Component: RDV inverters	RAW > 5.0 (CCF)	The RDV valves are 480 VAC motor operated valves which are powered from the 125 VDC class 1E vital buses via dedicated inverters. Failure of the inverters would result in failure of the RDVs to open. These inverters are continuously energized and failures are indicated. Maintenance is performed on failure.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Containment Spray System (CSS)	RAW > 5.0, Level 2	The CSS provides containment temperature and pressure control following accidents such as LOCAs and steam line breaks inside containment. The CSS also provides containment temperature and pressure control following a severe accident.
Component: Containment Spray pumps	Level 2	The CSS pumps deliver spray flow from the IRWST to the spray headers. The CSS pumps are assumed to be tested on a quarterly basis with unscheduled maintenance performed on failure. The SCS pumps can be used to backup the CSS pumps, and the CSS pumps can be used to backup the SCS pumps.
Component: Low Temperature Overpressure (LTOP) valves	Shutdown Risk Evaluation, RAW > 5.0	The LTOP valves are to provide RCS overpressure protection during low temperature operations. These valve also provide a "bleed" path for feed and bleed cooling during low temperature low pressure conditions with the shutdown cooling system valves open. It was assumed that these valves are tested and maintained consistent with current practices.
SYSTEM: Emergency Containment Spray Backup System (ECSBS)	Level 2	The function of the ECSBS is to provide an independent self contained means of supplying water to the containment spray header for containment heat removal during emergency conditions where the CSS and SCS pumps are not available.
Component: CS header "T"	Level 2	An 8 inch "T" connector is provided in each CS header line outside containment so that the CS headers can accept spray flow from an external source.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> ECSBS pumping device	Level 2	The ECSBS will have an independent pumping device that is capable of delivering sufficient flow to the containment spray headers at 24 hours after the onset of a severe accident to prevent the pressure in containment from exceeding level C pressure limits. No specific assumptions were made about the testing or maintenance of this pumping device.
<u>SYSTEM:</u>StartUp Feedwater System	Engineering Judgement	The function of the startup feedwater system is to provide feedwater flow during low power/startup/shutdown conditions. The startup feedwater system can backup the EFWS.
<u>Component:</u> Startup feedwater pump	Engineering Judgement	The startup feedwater pump is used to provide feedwater flow during low power/startup/shutdown conditions. The startup feedwater pump can act as a backup to the EFW pumps. Generic failure rates were used for this component. No specific assumptions were made as to the test interval or the maintenance frequency for this pump.
<u>SYSTEM:</u> Main Steam System	RAW > 5.0	The Steam Removal System consists of the main steam line and associated valves up to the turbine control valves. The valves in this system include those which provide for containment isolation following a steam line break or an SGTR and those which provide steam removal during a cooldown of the plant.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: Atmospheric Dump Valves (ADVs)	RAW > 5.0 (CCF)	These valves provide a controllable means of releasing steam from an SG to the atmosphere to prevent challenging the MSSVs. These valves were assumed to be tested on a quarterly basis.
Component: Main Steam Isolation Valves (MSIVs)	Engineering Judgement (CCF)	The MSIVs provide containment isolation following a steam line break. The MSIVs are also used to isolate the ruptured Steam Generator (SG) following an SGTR once the RCS pressure has been reduced to the point at which the MSSVs will not lift. The MSIVs have a partial stroke test on a quarterly basis with a full stroke test on a cold shutdown basis. Maintenance is performed only when the plant is shutdown.
Component: Main Steam Safety Valves (MSSVs)	Engineering Judgement	The MSSVs are the code safety valves for the SGs. Failure of the MSSVs to open could result in overpressurization of the SGs. If the MSSVs are challenged following an SGTR and fail to reseat, they provide a direct release path to atmosphere. The MSSVs are not tested at power. They were assumed to be tested consistent with current practices.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
<u>Component:</u> Turbine Bypass Valves (TBVs)	Engineering Judgement	Following a turbine trip, the turbine bypass valves open to discharge steam directly to the condenser, bypassing the turbine, to avoid unnecessary reactor trips and to prevent opening of the Primary Safety Valves and the Main Steam Safety Valves. The turbine bypass valves are air operated valves that fail closed on loss of air. These valves are interlocked so that they do not open on turbine trip if the condenser is not available. A generic failure rate was used for these valves. No specific assumption was made as to the test and maintenance intervals for these valves.
<u>SYSTEM:</u> Reactor Coolant System (RCS)		
<u>Component:</u> Primary Safety Valves (PSVs)	RAW > 5.0 (CCF)	The PSVs are the code safety valves for the RCS. Failure of one of these valves to reseat following a challenge such as an ATWS would result in LOCA to be mitigated. THE PSVs can not be tested at power. It was assumed that the PSVs are tested consistent with current practices. All PSV maintenance is performed while the plant is shutdown and depressurized.
<u>Component:</u> Reactor Coolant Pump (RCP) supports	SMA	Failure of the RCP supports during a seismic event could result in excessive RCP motion with the potential for failure of multiple RCS cold legs during the seismic event. Seismically induced failure of the RCP supports was the second dominant contributor to the plant HCLPF.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Chemical and Volume Control System (CVCS)	RAW > 5.0	The primary function of the CVCS is to provide RCS chemistry and volume control during normal power operation. The charging subsystem provides a mechanism for injection boron for long term reactivity control following an ATWS, and a mechanism for refilling the IRWST. The charging subsystem also provides the RCP seal injection function.
Component: Charging pumps	Engineering Judgement	The normal function of the charging pumps is to provide RCS volume control during normal operation. The charging pumps also provide RCP seal injection flow. The charging pumps can provide boron injection capability for long term reactivity control following an ATWS and can provide IRWST inventory makeup flow. During normal operation, one charging pump is running and the other is in standby. If the operating pump fails, the standby pump will be started, and unscheduled maintenance would be performed on the failed pump.
Component: Dedicated seal injection pump	Engineering Commitment	The dedicated seal injection pump is a positive displacement pump whose function is to provide seal injection flow to the RCP seals in the event that RCP seal cooling is unavailable due to the combined unavailability of the CCW/SSW system, concurrent with any charging pump failing to provide seal injection flow. The PRA did not include any specific test or maintenance assumptions for this pump.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Hydrogen Mitigation System (HMS)	Engineering Judgement (Level 2)	The function of the HMS is to burn the hydrogen released in containment during a severe accident in a controlled manner. The HMS consists of two redundant trains of hydrogen igniters. x% of the igniters in each train are powered from the station batteries and the rest are powered from the vital AC buses. The HMS is tested on a fuel cycle basis and maintenance is performed while the plant is shutdown.
Component: Hydrogen Igniters	Engineering Judgement (Level 2)	The hydrogen igniters are tested on a fuel cycle basis and maintenance is performed while the plant is shutdown.
SYSTEM: Cavity Flood System (CFS)	Engineering Judgement (Level 2)	In the event of a severe accident, the CFS provides a means of flooding the reactor cavity with water to cool the corium. The cavity is flooded from the IRWST by opening the holdup spillway valves from the IRWST to the HoldUp Tank (HUT) and by opening the cavity spillway valves from the HUT to the cavity.
Component: Holdup Spillway Valves	Engineering Judgement (Level 2)	The holdup spillway valves provide the means of flooding the HUT from the IRWST. The cavity is flooded from the HUT via the cavity spillway valves. The holdup spillway valves can not be tested at power because opening the valves would result in an unwanted flooding of the HUT. Therefore, the holdup spillway valves are tested on a fuel cycle basis when the plant is shutdown. Maintenance is performed on these valves only when the plant is shutdown

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: Cavity Spillway Valves	Engineering Judgement (Level 2)	The cavity spillway valves provide the path for flooding the cavity from the HUT, which is flooded from the IRWST via the holdup spillway valves. During normal power operation, the HUT is empty. Thus, in the PRA it was assumed that the cavity spillway valves could be stroke tested on a quarterly basis.
NUCLEAR ANNEX:		
Component: Fire Doors	Fire Risk Evaluation	Three hour rated fire doors are provided for each fire zone to prevent the propagation of fire from one fire zone to another. All fire doors are normally closed and their positions are indicated in the control room. If a fire door must be held open for maintenance access or other reason, a fire watch is maintained at the affected fire doors. No specific assumptions were made in the PRA as to the maintenance for the fire doors.
Component: Flood Doors	Flood Risk Evaluation	Flood doors are provided for each flood zone in the Nuclear Annex subsphere area to prevent the propagation of an internal flood from one flood zone to another. These flood doors are normally closed and their positions are indicated in the control room. If a flood door must be held open for maintenance access or other reason, a watch is maintained at the affected flood door. No specific assumptions were made in the PRA as to maintenance for the flood doors.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
SYSTEM: Containment		
Component: Containment shell	Level 2, SMA	The containment shell is the primary barrier preventing release of radioactive material following a core damage accident. Also, seismically induced overturning/sliding of the containment shell was found to be the dominant contributor to the plant level HCLPF value in the SMA.
Component: IRWST/HUT screens	Engineering commitment	The HUT/IRWST screens prevent trash and debris from entering the HUT and the IRWST and potentially blocking the suction lines for the safety pumps or the cavity flood valves. These screens are assumed to be inspected and cleaned on a fuel cycle basis. They are indirectly tested during the quarterly safety injection pump tests.

TABLE 19.15.6-1 RISK SIGNIFICANT SSCs FOR INCLUSION IN THE D-RAP		
SSC	RATIONALE FOR INCLUSION	INSIGHTS AND ASSUMPTIONS
Component: Containment penetration seals	Level 2	Containment integrity is important to reduce the risk to the public. The major containment penetrations (equipment hatch, personnel airlocks and fule transfer tube) will be designed to assure that they will not fail up to the ASME service level "C" pressure for the containment shell. 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. Containment failure from high temperatures due to a dry cavity contribute little to public risk in the System 80+ PRA. Consequently, the penetrations are not specifically designed to the low probability dry cavity scenario. However, to maximize containment integrity, the penetration design process will consider high quality and high capability seals as well as double seals (inner and outer) as applicable.

* For SSCs which contain (CCF) following "RAW > 5.0", the RAW is based on common cause failure of two or more of the specified SSC.

TABLE 19.15.6-2 Important Operator Actions from the PRA			
IMPORTANT OPERATOR ACTION	SOURCE	RAW*	RRW#
Operator fails to initiate Hot Leg Injection	Level 1	1.62E+02	1.02E+00
Operator fails to align EFWSTs to CST	Level 1	1.29E+02	1.01E+00
Operator fails to initiate RCS heat removal via Feed and Bleed Cooling	Level 1	1.93E+01	1.20E+00
Operator fails to initiate SCS for long-term decay heat removal	Level 1	6.81E+00	1.00E+00
Operator fails to align SCS pumps for RCS injection following aggressive secondary cooldown	Level 1	5.49E+00	1.02E+00
Operator fails to perform aggressive secondary cooldown (permitting injection via SCS) after SGTR	Level 1	3.49E+00	1.19E+00
Operator fails to perform aggressive secondary cooldown (permitting injection via SCS) after small LOCA	Level 1	2.85E+00	1.12E+00
Operator fails to reclose ADVs on ruptured SG	Level 1	1.98E+01	1.00E+00
Operator fails to align CVCS to refill IRWST following an SGTR	Level 1	4.07E+00	1.01E+00
Operators fail to align AAC to Vital AC buses following loss of offsite power and failure of EDGs	Level 1	Medium (Engineering Judgement	Medium (Engineering Judgement)
Operator fails to initiate cavity flooding	Level 2	High@	High@
Operator Fails to Depressurize RCS prior to Vessel breach using RDVs	Level 2	High @	High @
Operator fails to initiate Hydrogen Mitigation System	Level 2	Medium@	Medium@
Operator fails to align emergency backup containment spray system for use.	Level 2	Medium@	Medium@

TABLE 19.15.6-2 Important Operator Actions from the PRA			
IMPORTANT OPERATOR ACTION	SOURCE	RAW*	RRW#
Operator fails to Isolate an isolatable LOCA and isolate containment during shutdown operations	Shutdown Risk Evaluation	1.87E+02[1]	1.01E+00[1]
Operator fails to initiate feed using SCS during shutdown operations	Shutdown Risk Evaluation	2.21E+01[1]	1.42E+00[1]
Operator fails to start and load standby AC source during shutdown operations	Shutdown Risk Evaluation	1.20E+01[1]	1.40E+00[1]
Operator fails to isolate an Isolatable leak/LOCA during shutdown operations.	Shutdown Risk Evaluation	1.08E+01[1]	1.16E+00[1]
Operator fails to start standby SCS train during shutdown operation	Shutdown Risk Evaluation	8.56E+00[1]	1.32E+00[1]
Operator fails to initiate feed using the SIS during mode 5 operations	Shutdown Risk Evaluation	7.06E+00[1]	1.24E+00[1]
Operator fails to restore SCS train given that leak/LOCA is isolated	Shutdown Risk Evaluation	6.19E+00[1]	1.00E+00[1]
Operator fails to restore DHR in 73 hours given loss of DHR in mode 6 with refueling pool full and IRWST empty.	Shutdown Risk Evaluation	5.40E+00[1]	1.05E+00[1]
Operator fails to recover DHR within 12 hours following loss of DHR with successful boiloff makeup using the CVCS	Shutdown Risk Evaluation	4.15E+00[1]	1.54E+00[1]
Operator fails to use the CVCS to makeup inventory in mode 5	Shutdown Risk Evaluation	3.52E+00[1]	1.21E+00[1]
Operator fails to suppress fire during shutdown operations	Shutdown Risk Evaluation	2.76E+00[1]	1.24E+00[1]

NOTES FOR TABLE 19.15.6-2

* RAW = Risk Achievement Worth

RRW = Risk Reduction Worth

! The RAW and RRW values presented for the Shutdown Risk Evaluation are event tree branch point RAW and RRW values and include contributions from both operator errors and equipment failures.

@ The RAW and RRS values were not calculated for Operator actions in the level 2 analyses. Qualitative importances assigned based on engineering judgement.

19.15.~~6~~7 USE OF PRA TO SUPPORT CERTIFICATION ACTIVITIES

The System 80+ PRA results and insights are used in support of pre- and post-certification activities. The majority of the insights are identified during the pre-certification stage of the design. As a result, this has lead to further improvements in the design to eliminate or minimize potential vulnerabilities during the review process. The following activities include the use of PRA insights in support of design certification process.

A. Understanding of the design robustness to severe accidents - PRA insights are used to develop an in-depth understanding of the robustness and tolerance of the System 80+ design to severe accidents initiated by events which are either internal or external to the plant systems.

B. Importance of operator interface with the design - PRA insights are used to identify risk significant human errors associated with the System 80+ design. By characterizing the risk significant human error, new operating procedures can be developed or existing procedures refined to provide better training to plant operators.

C. Development and implementation of other programs - the PRA results and insights were used to systematically identify the key assumptions, major operator actions, and risk significant components that characterize the "present" risk of the System 80+ design. This information was used to support such programs as: (1) Design Acceptance Criteria (DAC), (2) Inspection, tests, analyses, and acceptance criteria (ITAAC), and (3) Reliability Assurance Program (RAP).

The PRA for the System 80+ design provides adequate models and associated data to effectively support the above mentioned certification activities.

ATTACHMENT II

The process used to produce CESSAR-DC provides a high degree of assurance that all information is correct and complete. While formal independent design verification was not performed in all cases additional extraordinary steps were taken consistent with the high degree of standardization and prior experience, to reduce the probability of errors being discovered at a later date.

Some of these considerations are described below:

The System 80+ design is an evolutionary design and is based heavily on the operating System 80 designs and the ABB-CE Korean designs under construction. The major components (Reactor Vessel, Fuel, Reactor Coolant Pumps) are identifical to System 80. Other components such as SG's, Pressurizer and Auxiliary Safety Systems such as the Shutdown Cooling and Safety Injection Systems have the same basic design. Some of the System 80+ systems that are not in System 80 (such as the Safety Depressurization System) are however in the ABB-CE Korean designs. It should be important to note that System 80 and the ABB-CE Korean designs have undergone complete design verification.

In the System 80+ design, the analytical results (e.g., for Chapters 6 and 15 safety analysis) were continually compared and evaluated against the System 80 and the Korean plants. Therefore, the expected results were undergoing technical verification as the design process was being followed.

The adequacy of the System 80+ design and analysis information was further verified by the following:

- CESSAR-DC Chapter Champions reviewed the data to confirm the data used in one chapter was not only consistent with the supplied data but was being used in the right manner. Also, this review provided the confirmation that the requirements imposed by design and results from one SAR chapter could be met by other design chapters. The computer codes used in performing the safety analyses were independently verified to assure that the methods were correct.

- Supervisors performed detailed technical review to confirm that the analyses and design results were correct. This was achieved by both the experience level of ABB-CE supervisors and the comparison with expected results based on the System 80 and Korean designs.

- Management Design Reviews for new designs, components or operations. These reviews required analytical results which again, in turn, were compared to the System 80 and Korean designs.

- Engineering Team Reviews were conducted to ensure consistency and verify technical adequacy based on System 80 and Korean designs.

- Subcontractors performed independent design verification of the safety related material supplied to ABB-CE. This was required because the ABB-CE scope did not cover this input in the System 80 and Korean designs.

ABB COMBUSTION ENGINEERING NUCLEAR SYSTEMS
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WINDSOR, CT 06095

ABB-CENP SYSTEM 80+ QUALITY ASSURANCE PLAN DESCRIPTION
PLAN NO. 18386-QO-001

LIST OF ACRONYMS

ABB-CENSYS	ABB Combustion Engineering Nuclear Systems
ALWR	Advanced Light Water Reactor
BOP	Balance of Plant
CESSAR	Combustion Engineering Standard Safety Analysis Report
DOE	Department of Energy
FDA	Final Design Approval
FY	Fiscal Year
NSSS	Nuclear Steam Supply System
PRA	Probabilistic Risk Analysis
QA	Quality Assurance
QPI	Quality Program Instructions
QPM	Quality Program Requirements Manual
QPR	Quality Program Requirements

1.0 PURPOSE AND SCOPE

This Quality Assurance Plan establishes the basic quality assurance requirements and approach governing activities performed by Combustion Engineering Nuclear Systems (ABB-CENSYS) for Design Certification of the System 80+ Advanced Light Water Reactor (ALWR). These efforts are provided in part under the Department of Energy (DOE) Contract No. 92791.

This QA Plan encompasses the current and long-term work scopes, and shall be maintained to reflect changes in these work scopes.

This QA Plan provides the basis for the organizational structure and procedures that shall be used to assure that ABB-CENSYS System 80+ Design Certification activities are performed in a controlled manner; that quality products are assured through the application of sound engineering standards, quality practices, and technical specifications; that supporting technology data are valid and retrievable; and that the program complies with contract requirements.

This ABB-CENSYS System 80+ QA Plan prescribes the policies, procedures, and instructions to be implemented to assure quality achievement in accordance with requirements contained in References 2.1 through 2.4; unless otherwise noted herein.

The purpose of this plan is to provide requirements for the execution of the DOE ALWR Design Certification Program and application for Design Certification of the System 80+ Standard Nuclear Power Plant Design. The Scope of the System 80+ Design is the complete Nuclear Power Plant engineered by ABB-CENSYS and designated subcontractors. The design process covered by this plan extends only to the point of obtaining NRC Certification and, therefore, does not include work necessary to be able to prepare procurement documents.

2.0 REFERENCES

2.1 ANSI/ASME NQA-1 (1986 Edition)

2.2 Description of Nuclear Power Businesses Quality Assurance Program, CENPD-210A, (See Note 1).

2.3 ABB Combustion Engineering Nuclear Systems Quality Program Requirements Manual, QPM-1. (See Note 1)

2.4 ABB Combustion Engineering Nuclear Systems Quality Program Instructions, QPM-1.1. (See Note 1)

NOTE 1: The latest revision applies unless otherwise specified.

3.0 SYSTEM 80+ QUALITY ASSURANCE PROGRAM REQUIREMENTS AND IMPLEMENTING PROCEDURES

These QA requirements are to be satisfied by implementing applicable portions of Reference 2.3 and Reference 2.4 and by implementing this QA Plan.

3.1 Organization

3.1.1 The organizational structure including the functional responsibilities, levels of authority and lines of communication for the control, maintenance and implementation of this QA Plan, is described in Reference 2.3 (QPM-1). Policies regarding the implementation of the ABB-CE Nuclear Systems Quality Assurance Program are established in the Policy section of QPM-1, Reference 2.3.

3.1.2 The ABB-CENSYS System 80+ project organization is shown in Figure 3-1. Figure 3-1 defines the organization elements which function under the cognizance of this QA Program and the lines of responsibility. The line of communication between ABB-CENSYS and the subcontractors is also shown.

3.1.2.1 The Design Certification Project Manager is responsible for the direction and integration of the efforts contributing to System 80+ design certification and also for the establishment, maintenance and implementation of the ABB-CENSYS System 80+ Quality Assurance Program as described in this QA Plan.

3.2 Quality Assurance Program

3.2.1 The ABB-CENSYS QA procedures related to this QA Plan are contained in Reference 2.4. Application of these procedures are as required unless otherwise specified herein.

3.3 Design Control

3.3.1 The ABB-CENSYS program and procedures related to Design Control are included in QPM-1 Sections 0300 and 0500, Reference 2.3 and QPM-1.1, QPIs 0301, 0302, 0303, 0305, 0307, 0308, 0313, 0401, 0701, 1701, Reference 2.4.

3.3.2 The goal of the Design Certification Program is certification of the System 80+ design by the Nuclear Regulatory Commission. The level of design detail to be produced is therefore defined as the information necessary to support NRC approval of CESSAR-DC.

The starting point for the System 80+ Design Certification program is the System 80 design as represented by CESSAR-F (which holds a Final Design Approval from the NRC in conformance with existing regulations) and the Duke Power Company P-81 BOP design. Design detail is available for System 80 NSSS and CESSAR-F, as they currently exist. The P-81 BOP design progressed to the PSAR stage before the project was canceled. Changes made to System 80 and CESSAR-F must be sufficiently detailed to support NRC review and to resolve safety issues.

CESSAR-DC will impose constraints on future contract design activities. QPM-1.1, QPI 0311, which governs SAR revisions, shall be applied to SAR material before a formal regulatory submittal is made. Other design control QA procedures in QPM-1.1, e.g., QPI 0304, which governs design analyses, and QPI 0306 governs design verification shall not be applied at this time. Such QA requirements will apply to the detailed design done after contract award for a System 80+ from a U.S. utility.

As part of the design certification effort, commitments will be made in CESSAR-DC which will become mandatory by the NRC's certification. Studies will be performed where necessary to provide the basis for the design described in CESSAR-DC.

In addition, where as-procured or as-built information that was provided previously in plant-specific FSARs is not available, design and performance criteria for selected equipment (e.g., safety grade pumps), will be provided, or the methods, procedures, and acceptance criteria for the analyses and tests that will demonstrate conformance with the assumptions of the safety analysis will be described.

3.3.3 To provide reasonable assurance that the System 80+ design information is correct and appropriate for intended purposes, the following will be performed:

3.3.3.1 Multi-disciplinary reviews (Integrated Reviews) will be conducted periodically by teams appointed by Project Management. Results of the reviews will be documented.

3.3.3.2 Design information provided to NRC will be checked for consistency among relavent CESSAR-DC chapters and among other referenced documents.

3.4 Procurement Document Control

3.4.1 There is no procurement of System 80+ hardware. Procurement of engineering services are as defined in Section 3.7.

3.5 Instructions, Procedures and Drawings (See Reference 2.4)

3.6 Document Control

3.6.1 A project file of applicable System 80+ design information shall be maintained by Project Management . The file shall have an index to all included documents.

3.6.2 Drawings other than sketches shall be maintained in accordance with Reference 2.4.

3.6.3 The Project file shall include documents produced by CENSYS as well as those provided to CENSYS by suppliers.

3.6.4 Submittal of the contents of the file to Quality Records is addressed in 3.17.

3.7 Control of Purchased Items and Services

3.7.1 The Project Manager shall be responsible to provide direction to suppliers of engineering services.

3.7.2 Subcontracted services may be performed as safety-related or non-safety related.

3.7.3 The Project Manager shall ascertain the extent of QA Controls being implemented by suppliers and may augment controls over such work as desired (e.g. QA surveys or audits).

3.7.4 Prior to NRC issuance of a FSER, suppliers will be audited to verify that engineering services work was adequately controlled.

3.8 Identification and Control of Items - Not Applicable

3.9 Control of Special Processes - Not Applicable

3.10 Inspection - Not Applicable

3.11 Test Control - Not Applicable

3.12 Control of Measuring and Test Equipment - Not Applicable

3.13 Handling, Storage and Shipping - Not Applicable

3.14 Inspection, Test and Operating Status - Not Applicable

3.15 Control of Nonconforming Items - Not Applicable

3.16 Corrective Action (See Reference 2.4)

3.17 Quality Assurance Records

3.17.1 During execution of the Certification Program, design documents (including CESSAR-DC) may be designated as non-permanent or permanent quality records in accordance with QPI 1701 of Reference 2.4.

3.17.2 Within ninety (90) days of NRC issuing the FDA all design documents in the Project file that were not already designated quality records shall be so designated.

3.17.3 This QA Plan shall be designated as a non-permanent quality record.

3.18 Audits

3.18.1 In addition to Internal audits conducted in accordance with QPI 1801 of Reference 2.4, an audit to verify compliance with this QA plan shall be performed at least once prior to issuance of an FSER.

3.18.2 Supplier audits prescribed in 3.7.4 shall be conducted in accordance with QPI 1802 of Reference 2.4.

4.0 DELIVERABLES

4.1 The deliverables for which ABB-CE Nuclear Systems currently has responsibility and which comes under this QA Plan are: CESSAR-DC and the applicable submittal documents.

5.0 SPECIAL TRAINING REQUIREMENTS

5.1 System 80+ Project Management personnel, the Project Engineering Manager, and chapter champions (Per Fig. 3-1) shall have documented training, via self-study in this QA plan.

6.0 APPROVAL OF THE QA PLAN

6.1 This QA Plan shall be approved by the Vice President, Nuclear Systems Engineering, and the Director, Quality Assurance.

7.0 DISTRIBUTION AND USE OF THE QA PLAN

7.1 This document shall be used for System 80+ Design Certification, until it is superseded by a revised or new plan.

7.2 This QA Plan shall be distributed to those individuals directing System 80+ work affecting quality. A distribution list shall be maintained by the Project Manager.

8.0 REVISIONS OF THE QA PLAN

Revisions to this document shall be incorporated as required and the document will receive the same review, approval and distribution as the original issue. The revised document shall be issued in its entirety with the Record of Revisions Page identifying the affected sections.

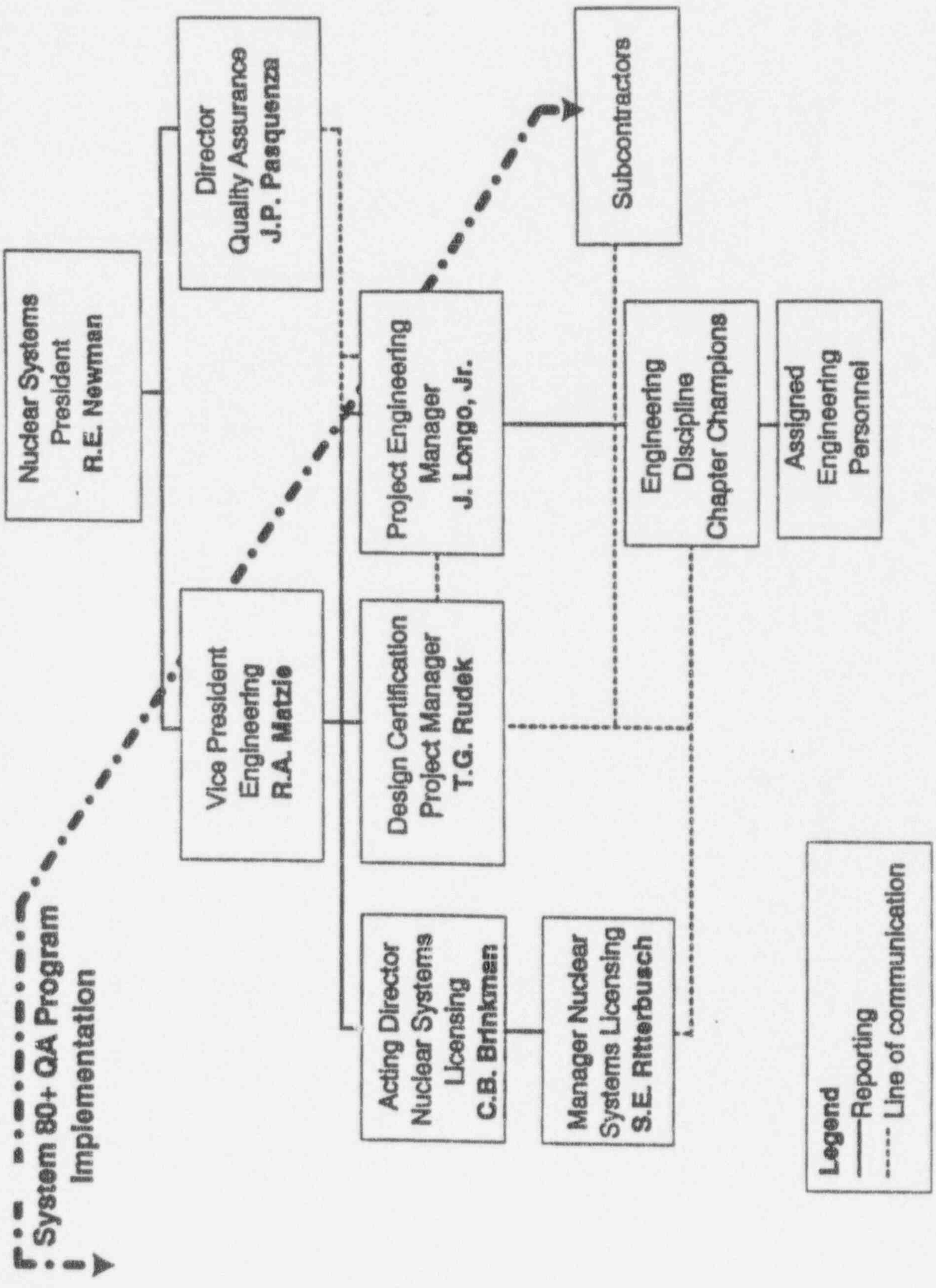

Figure 3-1

ABB Combustion Engineering Nuclear Systems
System 80+ Project Organization