ABB

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ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

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Mr. Douglas Coe
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Subject: ABB-CE Responses to ACRS Questions on System 80+™

Dear Mr. Coe:

This letter provides responses to questions (Enclosure I) raised by various members of the Advisory Committee on Reactor Safeguards (ACRS) ABB-CE Standard Plant Designs Subcommittee at the meeting held March 8-9, 1994. Also provided (Enclosure II) are revised responses to questions from previous ACRS ABB-CE Standard Plant Designs Subcommittee meetings. I believe that the attached responses will clarify the issues raised by members of the Subcommittee.

If I can be of further assistance regarding these matters, please do not hesitate to call me, or Mr. Stan Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

C. B. Brinkman, Director
Nuclear Systems Licensing

Enclosures: As stated

cc: T. Wambach (NRC)
P. Lang (DOE)

Enclosure I to LD-94-023

ABB Combustion Engineering

System 80+™ Standard Plant Design

Responses to ACRS ABB-CE Standard Plant Designs Subcommittee Questions
(March 8-9, 1994)

Responses to ACRS ABB-CE Standard Plant Designs Subcommittee Questions

(March 8-9, 1994)

Question 940308-1:

To what extent will the CE-SSAR clarify the hydrostatic pressure rating for seals around pipes which penetrate structural walls below the maximum probable flood level, specify the type of seal to be used, and specify the location of the penetration seals (e.g., state which of those pipes are located in tunnels which may interconnect separate buildings)? Will the maximum expected leakage rate upon seal failure be less than the sump pump capacity in the flooded location?

Response:

The System 80+ Nuclear Annex design incorporates a divisional wall such that flooding (internal and external) in one division can not impact the other division. Below grade penetrations to safety-related structures will either be directly penetrating piping, conduits or tunnels/enclosures. In all cases, the sealing material will be provided to preclude the entrance of water from either internal or external sources. Additionally, in the event of a seal failure, the credible leakage rate will be limited to the capacity of the sump pumps or the affects of the flooding will be shown to be acceptable.

CESSAR-DC Section 3.4.4.1 is being revised to state that the water seals are designed for the static pressure of water at the flood elevation. Hydrostatic loads are also addressed in CESSAR-DC Section 3.8A-5.1.1.4.

The connections to the Annex are discussed below:

1. Buried Cable Tunnels and Conduit Banks

   Appendix 3.8A Section 11.8 provides the criteria for buried tunnels and conduit banks. Section 3.8A-11.8 .6 is being revised to identify the design requirements for hydrostatic loads. Hydrostatic loads are also addressed in CESSAR-DC Section 3.8A-5.1.1.4.

2. Component Cooling Water Piping Tunnel

   This tunnel is addressed in Section 3.8.4.1.5 and Appendix 3.8A-11.7. Section 3.8A-11.7.7 is being revised to identify the design requirements for hydrostatic loads.

Response (continued):

3. Diesel Fuel Piping

   Diesel fuel piping is addressed in response to Question 940308-3.

4. Radwaste Building Piping to Annex

   This pipe chase is shown on Figures 1.2-5A and 1.2-5B. The sealing requirements of Section 3.4.4.1 apply to this tunnel.

5. Station Service Water Piping

   The CCW heat exchanger structure and related piping are discussed in CESSAR-DC Section 9.2.2.1.4. The service water piping only enters the CCW heat exchanger structure and does not penetrate the walls of the Nuclear Annex. As annotated in the marked up text and inserts to CESSAR-DC Section 3.8A-11.7.7-1 the flooding issues related to the service water pipe will be limited to the CCW heat exchanger structure and not impact the Nuclear Annex.

   The requirements for flood protection are identified in CESSAR-DC Section 3.4.4.1.

6. Other

   The connections to the Nuclear Annex are the piping connections to yard tanks (Holdup, Reactor Makeup, Boric Acid Storage, Condensate Storage). The requirements in Section 3.4.4.1 will be met for all penetrations into Seismic Category I structures.

Seismic Category I structures identified in Table 3.2-1 are designed for flood protection.

These Seismic Category I structures are designed to protect safety-related equipment from floods by incorporating the following safeguards into their construction:

A. No exterior access openings will be lower than 1 foot above plant grade (yard grade) elevation.

B. The finished yard grade adjacent to the safety-related structures will be maintained at least 1 foot below the ground floor elevation, except where ramps or steps are provided for access.

C. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation.

D. Water seals are provided for all penetrations in exterior walls up to flood level elevation. INSERT 3.4.4.1-1

For other Seismic Category I structures where flood protection measures are required (e.g. pumping systems, stoplogs, watertight doors, dikes, retaining walls and drainage systems) the design of means for providing such protection will be described in Section 2.4 of the site-specific SAR. Penetrations located below the external flood level in the external walls of the Nuclear Annex are currently projected to include Component Cooling Water, Radwaste, and Diesel Fuel Oil System piping and cable penetrations. Additional penetrations may be identified once layouts are finalized for systems such as sewage, demineralized water, station air, and security. INSERT 3.4.4.1-2

The COL applicant will perform an evaluation to ensure that all penetrations in Seismic Category I structures ~~below the external flood level~~ are properly sealed to protect safety related equipment from flooding.

External flooding as a result of secondary flooding sources located in the Turbine Building are addressed in Section 10.4.1.3. Entrances to the Nuclear Annex from the Turbine Building are elevated above plant grade to prevent flood propagation.

Internal flood protection in the System 80+™ design minimizes possible flood sources. The station service water system is located outside the Nuclear Annex to eliminate a significant source of water. The component cooling water and emergency feedwater systems are fully separated by division, thus eliminating the possibility of a single flood source within these systems impacting both divisions.

Lengths of high energy and moderate energy piping have been minimized by equipment location. Equipment in the Reactor Building (RB) Subsphere is located in quadrants to minimize the

Insert 3.4.4.1-1

The water seals are designed for the static pressure of water at the flood elevation. Water seals at the interface with safety related structures are designed to maintain integrity in the event of a Safe Shutdown Earthquake. In the event of seal failure, any credible leakage is limited to the capacity of the sump pumps or associated flood effects are shown to be acceptable.

Insert 3.4.1-2

All penetrations are sealed on the inside of the penetration to eliminate the potential of flooding through the penetration.

Insert 3.8A11.8.6-1

Cable tunnels and conduit banks shall be sealed against the introduction of exterior water sources into the tunnel or bank and shall be sealed at the interface with safety related structures to prevent flooding effects. The water seals are designed for the static pressure of water at the flood elevation. Water seals at the interface with safety related structures are designed to maintain integrity in the event of a Safe Shutdown Earthquake. In the event of seal failure, any credible leakage is limited to the capacity of the sump pumps in the safety related structure or the associated flooding effects are shown to be acceptable.

Insert 3.8A11.7.7-1

The Component Cooling Water Heat Exchanger Piping tunnel shall be sealed against the introduction of exterior water sources into the tunnel and shall be sealed at the interface with safety related structures to prevent flooding effects. The water seals are designed for the static pressure of water at the flood elevation. Water seals to preclude flooding of the Nuclear Annex caused by Service Water piping failure in the Component Cooling Water Heat Exchanger Structure and those at the interface to the Nuclear Island Structure are designed to maintain integrity in the event of a Safe Shutdown Earthquake. In the event of seal failure, any credible leakage is limited to the capacity of the sump pumps in the safety related structure.

movement shall be considered. Friction between the tunnel and the surrounding soil shall be considered using conservative estimates of the associated frictional forces.

11.7.5.4 Other Loads

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

11.7.6 LOADING COMBINATIONS AND ACCEPTANCE CRITERIA

11.7.6.1 Concrete

The tunnel shall be designed using Seismic Category I criteria. The requirements of Sections 5.2.2 and 8.0 of this appendix shall be met.

11.7.6.2 Stability

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

11.7.7 OTHER REQUIREMENTS

INSERT 3.8A. 11.7.7-1

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

11.8 BURIED CABLE TUNNELS, AND CONDUIT BANKS

11.8.1 CONDUIT CLASSIFICATION

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

11.8.2 DESCRIPTION

Buried cable tunnels and conduit banks are reinforced concrete box type structures, generally rectangular in cross-section that house conduit for electrical distribution.

11.8.3 CODES AND STANDARDS

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

11.8.4 LOADS

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

11.8.4.1 **Dead Load (D)**

The weight of the contents of the cable tunnel and/or conduit bank.

11.8.4.2 **Live Load (L)**

The structure shall be designed for soil overburden pressure, AASHO H20-44 truck loading and construction equipment loading as applicable to the specific site.

11.8.4.3 **Seismic Loads (E')**

The reinforced concrete buried cable tunnels and/or conduit banks shall be seismically designed to sustain soil movement during earthquake ground motions. The structural integrity of the cable tunnel and/or conduit bank is evaluated by accounting for the two primary effects of earthquake motion, namely;

1. Strains and associated stresses induced in the tunnel by the free-field vibration resulting from motions of the surrounding soil mass (seismic wave passage), and

2. Seismically induced differential movements of the ends of the tunnel (i.e., the Nuclear Island and the CCW Heat Exchanger Structure).

Equivalent static analysis shall be performed considering the conduit tunnel as a beam on an elastic foundation. Axial stress caused by seismic waves, soil friction, thermal expansion and differential movement shall be considered. Friction between the tunnel and the surrounding soil shall be considered using conservative estimates of the associated frictional forces.

11.8.4.4 **Other Loads**

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

11.8.5 LOADING COMBINATIONS AND ACCEPTANCE CRITERIA

11.8.5.1 **Concrete**

The cable tunnels and conduit banks shall be designed using Seismic Category I criteria.

11.8.5.2 **Stability**

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

11.8.6 OTHER REQUIREMENTS

The cable tunnels and conduit banks are to be founded on competent structural backfill as defined in Section 10.1 of this appendix. The bearing pressure shall not exceed the value given in Table 2.0-1.

INSERT 3.8 11.8.6-1

Insert 3.8A.11.8.6-1

Cable tunnels and conduit banks shall be sealed against the introduction of exterior water sources into the tunnel or bank and shall be sealed at the interface with safety related structures to prevent flooding effects. The water seals are designed for the static pressure of water at the flood elevation. Water seals at the interface with safety related structures are designed to maintain integrity in the event of a Safe Shutdown Earthquake. In the event of seal failure, the leakage is limited to the capacity of the sump pumps in the safety related structure.

Insert 3.8A.11.7.7-1

The Component Cooling Water Heat Exchanger Piping tunnel shall be sealed against the introduction of exterior water sources into the tunnel and shall be sealed at the interface with safety related structures to prevent flooding effects. The water seals are designed for the static pressure of water at the flood elevation. Water seals at the interface with the Nuclear Island and Component Cooling Water Heat Exchanger structure are designed to maintain integrity in the event of a Safe Shutdown Earthquake. In the event of seal failure, the leakage is limited to the capacity of the sump pumps in the safety related structure.

Question 940308-2:

The CE-SSAR does not appear to specify equipment (other than structural walls), such as doors, penetration seals, and ventilation dampers, which must be designed to withstand design tornado wind loading. What assurance is there that such equipment will be designed to meet this loading?

Response:

Exterior accesses to safety related structures are designed for the tornado pressures. CESSAR-DC Section 3.3 is being revised to clarify that accesses exposed to wind on Seismic Category I structures are designed for wind and tornado loadings. Dampers to protect against tornados are indicated on HVAC figures in Section 9.4.

3.3 WIND AND TORNADO LOADINGS

INCLUDING ACCESSES

All Seismic Category I structures, except those not exposed to wind, are designed for wind and tornado loadings.

3.3.1 WIND LOADINGS

The design for wind loading is in accordance with ANSI/ASCE 7, "Minimum Design Loads for Buildings and Other Structures" (Reference 1). Structural geometries not addressed in ANSI/ASCE 7 shall be evaluated using ASCE Paper 3269, "Wind Forces on Structures" (Reference 2), and ASCE Paper 4933, "Wind Loads on Dome-Cylinder and Dome-Cone Shapes" (Reference 3).

3.3.1.1 Design Wind Velocity

A design wind velocity of 110 mph, at a height of 33 feet above nominal ground elevation is used as the maximum wind speed for a 50 year recurrence period.

Velocity profiles and associated effective pressures for winds with a 100 year recurrence period are calculated in accordance with Section 6 of Reference 1 utilizing an Importance Factor, I, of 1.11 and Exposure C.

Gust response factors are dependent on height above grade level and are in accordance with Table 8 of Reference 1 for Exposure C.

3.3.1.2 Determination of Applied Forces

Based on structure geometry and physical configuration, the effective pressure distribution is transformed into applied equivalent static building forces utilizing appropriate shape coefficients given in Reference 3.

Wind pressure distribution curves for the containment shield building are shown in Figure 3.3-1. The maximum height of the shield building above grade is approximately 173 feet 3 inches.

3.3.2 TORNADO LOADINGS

All Seismic Category I structures, that perform a safe shutdown or accident mitigation function except those structures not exposed to wind, are designed for tornado loadings.

3.3.2.1 Applicable Design Parameters

Tornado effects are in accordance with Interim Regulatory Guide 1.76 (Reference 4). The following parameters are applicable to the design basis tornado.

Question 940308-3:

Please clarify the location of the gravity feed fuel oil piping running to the emergency diesel generators (i.e., buried or in a pipe chase or tunnel).

Response:

The design of the diesel fuel oil system and structure are described in CESSAR-DC Sections 1.2.16.7, 3.8.4.1.4, and 9.5.4. The building arrangement is shown on Figure 1.2-24. The piping will exit the fuel storage structure underground. The safety related piping will be routed to the Nuclear Annex/Diesel Generator Area within a safety related enclosure/pipe chase which is designed to Seismic Category I criteria and is protected from tornado-generated missiles.

The chase/enclosure will be designed to preclude the entrance of ground/flood water. The chase/enclosure will also protect the piping OD from the effects of soil corrosion, provide for a "conduit" to collect and direct any oil leakage to a suitable and observable location, and where required due to specific site conditions, provide the necessary external missile protection.

Question 940308-4:

What are the CE-SSAR requirements for the seismic qualification for underground pipe chases or tunnels which may contain non-seismic piping? How will the CE-SSAR address the potential for relative building motion during a seismic event, which may cause damage to piping which penetrates two adjacent buildings or its surrounding tunnel/chase?

Response:

All piping that has the potential to affect Seismic Category I components is designed as either Seismic Category I or II piping. CESSAR-DC Appendix 3.9A, Section 1.1 states that Seismic Category II piping is analyzed to ensure that the SSE does not adversely impact safety related equipment or components. This includes piping in tunnels/enclosures.

Potential relative building motion is considered as seismic anchor motion for piping analysis and design. CESSAR-DC Appendix 3.9A, Section 1.2.4 requires that the seismic anchor movements be considered.

The pipe tunnels/chases are also designed for the relative movement of the structures. CESSAR-DC Appendix 3.8A, Sections 11.7.5.3 and 11.8.4.3 identify the seismic effects for which the tunnels/chases must be designed, including the differential movements of the end of the tunnels.

Question 940308-5:

Please explain the apparent opening to the EDG room opening on CE-SSAR Figure 1.2-5A that does not show a door.

Response:

The opening shown on CESSAR-DC Figure 1.2-5A for the Emergency Diesel Generator room is for equipment removal and maintenance. A roll-up door is installed in this opening and is shown as two lines on the Diesel Generator room side of the opening. This roll-up door has a 3-hour fire rating as indicated on CESSAR-DC Figure 9.5.1-3.

Question 940308-6:

Regarding the divisionalized safety-related sumps which are pumped to a common header in the rad waste building: How many backflow prevention devices (e.g., check valves) would have to fail in order to create a flow path between divisionalized sumps?

Response:

Each of the divisionalized pipe headers from the safety-related sump pumps are provided with a Seismic Category I check valve at the penetration from the Nuclear Annex to the Radwaste Building pipe tunnel. The Radwaste Building pipe tunnel is at elevation 70+0. Since this penetration is through an exterior wall flood barrier of a Seismic Category I structure, a portion of the non-safety piping from the flood barrier through an interior valve for flood protection shall be designed to Seismic Category I criteria. Since the flow through these lines is out of the building a check valve shall be used for flood protection.

In addition, a reverse flow check valve is located at each of the sump pump discharges. Thus, there are two check valves in series which prevent back flow from the common header located in the Radwaste Building.

The check valves on the Radwaste Building Subsphere floor drain sump pumps are Safety Class 3. These valves are reverse flow tested quarterly as specified in the IST plan presented in CESSAR-DC Table 3.9-15. The check valves on the discharge of the Nuclear Annex radioactive floor drain sumps and CVCS area floor drain sumps are non-safety related. These check valves along with the Seismic Category I check valves at the flood barrier wall will be reverse flow tested quarterly as specified in interface requirements to the O-RAP.

If these check valves should fail the redundant sump pumps associated with the sump will automatically start on high level in the sump and prevent the sump from over flowing and thus flooding the adjacent areas.

To ensure that all non-safety related lines which penetrate a Seismic Category I exterior flood barrier are identified after they are routed and are designed to Seismic Category I criteria to an interior isolation valve the following is being added to Section 3.4:

> Non-safety related piping that penetrates an exterior wall flood barrier of a Seismic Category I structure is designed to Seismic Category I criteria from the wall through an interior isolation valve. For piping in which flow is exiting the structure the interior valve is a reverse flow check valve. For piping in which flow is entering the structure the valve is an isolation valve which can be manually isolated to terminate a break in the non-seismic portion of the interior piping to prevent flooding.

Seismic Category I structures identified in Table 3.2-1 are designed for flood protection.

These Seismic Category I structures are designed to protect safety-related equipment from floods by incorporating the following safeguards into their construction:

A. No exterior access openings will be lower than 1 foot above plant grade (yard grade) elevation.

B. The finished yard grade adjacent to the safety-related structures will be maintained at least 1 foot below the ground floor elevation, except where ramps or steps are provided for access.

C. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation.

D. Water seals are provided for all penetrations in exterior walls up to flood level elevation. (INSERT 3.4.4.1-1)

For other Seismic Category I structures where flood protection measures are required (e.g. pumping systems, stoplogs, watertight doors, dikes, retaining walls and drainage systems) the design of means for providing such protection will be described in Section 2.4 of the site-specific SAR. Penetrations located below the external flood level in the external walls of the Nuclear Annex are currently projected to include Component Cooling Water, Radwaste, and Diesel Fuel Oil System piping and cable penetrations. Additional penetrations may be identified once layouts are finalized for systems such as sewage, demineralized water, station air, and security. (INSERT 3.4.4.1-2)

INSERT 3.4.4.1-3

The COL applicant will perform an evaluation ~~to ensure that all penetrations in Seismic Category I structures below the external flood level~~ are properly sealed to protect safety related equipment from flooding.

External flooding as a result of secondary flooding sources located in the Turbine Building are addressed in Section 10.4.1.3. Entrances to the Nuclear Annex from the Turbine Building are elevated above plant grade to prevent flood propagation.

Internal flood protection in the System 80+™ design minimizes possible flood sources. The station service water system is located outside the Nuclear Annex to eliminate a significant source of water. The component cooling water and emergency feedwater systems are fully separated by division, thus eliminating the possibility of a single flood source within these systems impacting both divisions.

Lengths of high energy and moderate energy piping have been minimized by equipment location. Equipment in the Reactor Building (RB) subsphere is located in quadrants to minimize the

INSERT 3.4.4.1-3

~~Insert 3:~~

Non-safety related piping that penetrates an exterior wall flood barrier of a Seismic Category I structure is designed to Seismic Category I criteria from the wall through an interior isolation valve. For piping in which flow is exiting the structure the interior valve is a reverse flow check valve. For piping in which flow is entering the structure the valve is an isolation valve which can be manually isolated to terminate a break in the non-seismic portion of the interior piping to prevent flooding.

Question 940308-7:

Please explain your rationale for specifying Class II seismic piping for emergency diesel generator fire suppression piping, instead of Class I.

Response:

The preaction sprinkler system piping and sprinkler heads inside the diesel generator rooms will be designed to Seismic Category I requirements. The following is being added to CESSAR-DC Section 9.5.1.7.3.C, item 1(also see attached CESSAR-DC page mark-up);

"The preaction sprinkler system piping and sprinkler heads inside the diesel generator rooms are designed to Seismic Category I requirements."

C. Systems Interaction

1. Sprinkler system piping is seismically restrained to avoid interaction with systems, equipment, and components which must function following the design basis seismic event. ← add Insert A

2. Sprinkler head locations are selected and analyzed to assure that water spray does not expose redundant equipment required to achieve cold shutdown or high voltage electrical equipment which may result in a personnel hazard.

3. Sprinkler systems are analyzed to assure that pipe break/water spray does not potentially expose redundant equipment required for cold shutdown.

4. Sprinkler system drains and test connections are routed to unit drains to control water discharge.

5. In areas where equipment is subject to damage by water accumulation, floor drains are provided or equipment is installed on elevated platforms to avoid damage.

6. Sprinkler heads are located as required by NFPA 13. Other plant equipment and components are located so that they do not obstruct the designed sprinkler water discharge pattern. If obstruction is unavoidable, additional sprinkler heads are installed to assure proper water distribution.

7. Where installed, automatic sprinkler systems are considered primary protection. Portable extinguisher and fire hose stations are provided for back-up protection.

CESSAR-DC Section 3.4.4.1 contains a discussion of internal flood protection methods. These flood protection methods will protect safe shutdown equipment from internal flooding including flooding due to water released during fire suppression activities.

Sprinkler system components including manual isolation valves, preaction control valves, pipe, fittings, hangers, sprinkler heads, and detectors are Underwriter's Laboratories Listed or Factory Mutual Approved for use in fire protection systems. An exception to Listed or Approved equipment is containment isolation valves which are not available as Listed or Approved.

Insert A:

The preaction sprinkler system piping and sprinkler heads inside the diesel generator rooms are qualified to Seismic Category I requirements.

Question 940308-8:

On CE-SSAR page 3.6-3 your definition of a high energy system includes the requirement that it is pressurized above atmospheric during normal plant operation. This appears to make the system energy classification dependent upon the plant operating mode. Is this correct?

Response:

Appendix A to Branch Technical Position SPLB 3-1, Rev. 2 - October 1990 has the following definition for High-Energy Fluid Systems:

> Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where both of the following are met:
>
> a. maximum operating temperature exceeds 200°F, or
>
> b. maximum operating pressure is 275 psig or less

Section 3.6.1.1.1 of CESSAR-DC is being revised to remove the words "above atmospheric pressure" and the words "normal plant operations" is being revised to "normal plant conditions".

High- and moderate-energy pipe failure locations are postulated as described in Section 3.6.2. Each postulated rupture location is evaluated for its effect on safe shutdown systems and components required following the specific pipe failure event.

3.6.1.1.1 High-Energy Piping Systems

A high-energy pipe failure is postulated in branches or piping runs larger than one inch nominal diameter and which operate during normal plant conditions with high energy fluid.

Included in this category are fluid systems or portions of fluid systems which are pressurized ~~above atmospheric pressure~~ during normal plant operation and which, in addition, operate during normal plant conditions and where either or both of the following are met:

A. Maximum operating temperature exceeds 200°F, or

B. Maximum operating pressure exceeds 275 psig.

Fluid piping systems that qualify as high-energy for only short portions of their operational period are considered moderate-energy systems if the portion of their operational period within the pressure and/or temperature specified above for high energy fluid systems is less than two percent of the time period required to accomplish its system design function.

In analyzing the effects of a high-energy pipe failure, the consequences of pipe whip, water spray, jet impingement, flooding, compartment pressurization, and environmental conditions are considered.

See Appendix 3.9A, Section 1.1.8.1.1 for a further discussion.

3.6.1.1.2 Moderate-Energy Piping Systems

A moderate-energy pipe failure is postulated in branches or piping runs larger than one inch nominal diameter and which operate during normal plant conditions with moderate-energy fluid.

Included in this category are fluid systems or portions of fluid systems which are pressurized above atmospheric pressure during normal plant operation and which, in addition, operate during normal plant conditions and where both of the following are met:

A. Maximum operating temperature is 200°F or less, and

B. Maximum operating pressure is 275 psig or less.

Question 940308-9:

Please provide the topical report, referenced in Chapter 6, which analyzes the HELB pressurization of the main steam valve house to 10 psi. What requirements will be specified in the CE-SSAR for the qualification of doors and seals?

Response:

A copy of the Topical Report CENPD-141, Revision 2, "A Description of the DDIFF-1 Digital Computer Code for Reactor Plant Subcompartment Analysis", is attached. This report is Reference 19 in CESSAR-DC Section 3.6.

In Seismic Category I (safety-related) structures doors are designed for subcompartment pressures. CESSAR-DC Section 3.8.4.3.I identifies subcompartment pressures and temperatures as loading conditions. A statement is being added to Section 3.8.4.3.I to clarify that doors are designed for the subcompartment pressures.

I. Pressure and Temperature Loads

The Seismic Category I structures are designed for global effects of pressure, if any, and temperature gradients, in addition to accident temperature gradients as a result of postulated pipe ruptures described in Section 3.6.1. Exterior walls and roofs above plant finished yard grade are designed for the 1 percent exceedance ambient temperature values in Table 2.0-1.

The potential for global temperature and pressure loads in the Nuclear Annex is minimized by the selected routing of high pressure lines as described in Section 3.6.1.1.

Nuclear Annex subcompartments susceptible to potential global pressure and temperature and the design basis for these effects are as follows:

a. The Main Steam Valve House (MSVH) is designed for a 10 psi differential pressure across the walls as a result of a Main Steam line break. The MSVH temperature is 300°F. The short duration of this temperature results in a negligible thermal load across the MSVH wall.

b. The pipe chase containing the Chemical and Volume Control System (CVCS) 2 inch letdown line is designed for compartmental pressure and temperature resulting from a postulated rupture of this line.

The Shield Building annulus global temperature and pressure loads from postulated events are described in Section 6.2.1.8. The values identified are 9.3 inches of water and 212°F which are applied for an extended period of time. The Shield Building annulus wall pressure design is governed by a tornado event which results in a differential pressure across the Shield Building wall of approximately 4 psi.

2.4

INSERT TP 3.8.4.3.I-1

Loading combinations used for the design of Seismic Category I structures are shown in Table 3.8-5 and Appendix 3.8A.

3.8.4.4 <u>Design and Analysis Procedures</u>

Seismic Category I and II concrete and steel structures, with exception of the steel containment vessel, are designed in accordance with the criteria in Appendix 3.8A.

The Seismic Category I structures are designed to prevent possible overturning, sliding and flotation. The forces and moments acting on the building which could cause these events are determined for the different loads and load combinations and are then compared to the corresponding forces and moments which resist overturning, sliding or flotation. Safety factors for the

Insert 3.8.4.3.I-1

Access doors are designed for the subcompartment pressures when there is a potential to effect safety related equipment if the door fails to retain the pressure boundary.

CENPD-141

Revision 2

DDIFF-1 CODE

A Description of the DDIFF-1
Digital Computer Code for
Reactor Plant Subcompartment Analysis

Plant Engineering

March, 1978

Combustion Engineering, Inc.
Nuclear Power Systems
Windsor, Connecticut 06095

Question 940308-10:

Please summarize the hand calculation for compartment pressurization due to rupture of the EFW steam line.

Response:

A hand calculation was performed assuming steam flow into the EFW pump room from the 6 inch line to the EFW pump turbine. The flow from the broken line was assumed to be the critical flow value based on a constant source pressure of 1100 psia conservatively neglecting the line loss from the main steam line to the EFW pump room.

The pressure for the EFW pump room was obtained in the following way. A hand calculation was performed to determine the steady state pressure which the room would achieve. The flow resistances through the pipe chase between the EFW pump room and the main steam valve house (MSVH) and the resistances between the MSVH and the environment were calculated. Using the Darcy equation, the pressure in the EFW pump room required to match the steam flow from the broken line was determined. This calculation resulted in a EFW pump room pressure of about 2 psi. The design pressure of the EFW pump room was conservatively chosen to be 10 psi.

A confirmatory DDIFF-1 (Reference 1, copy attached to response to previous question) analysis of this event was also performed. The result was a peak pressure of 3.5 psi. The difference in pressures between the hand calculation and DDIFF-1 is the result of effects of terms such as inertia. The 3.5 psi pressure is well below the design pressure of 10 psi.

Reference:

1. CENPD-141 Revision 2, "A Description of the DDIFF-1 Digital Computer Code for Reactor Plant Subcompartment Analysis", March, 1978.

Question 940308-11:

Where is the COL action item specified for performing plant walkdowns to verify the assumptions of the high energy line break analyses?

Response:

CESSAR-DC Section 3.6.2 requires that an inspection of the as-built high energy pipe break features be performed and that final designs and results of high and moderate energy piping analyses be documented in a pipe break analysis report. Table 3.1-1, Item 4 in the Certified Design Material further documents the requirement for this report.

Question 940308-12:

Please describe the basis for your position that there is a low probability of a control room fire. Include any specification of the amount and type of electrical wiring to be used.

Response:

The Nuplex 80+ control room includes features to reduce the probability that a fire will start. The major concern is the control panel materials and control panel wiring. CESSAR-DC Section 7.7.1.3.1 requires use of fire retardant material throughout the control panel enclosures(e.g., meeting UL-94 rating). Non-metallic materials used in control panels will neither ignite nor explode and will not independently support combustion. Fire resistant insulation material for control panel wiring must meet the applicable requirements of IEEE Standard 383 per CESSAR-DC Section 7.7.1.3.1.

Energy sources entering the control panels are limited to 125V. Approximately 95% of the power distribution cable and wiring within the panels is low voltage (5Vdc-24Vdc), practically eliminating potential ignition sources. Within the control panels, power wiring is run separately from the low voltage wiring. A reduced number of panel devices and the low voltage interfaces and lamp LEDs also contribute to reducing fire probability. Each panel section is provided with high temperature switches to provide indication of abnormal operating temperatures and smoke detectors to indicate internal fires. Power can be removed from a control room panel section affected by fire by channelized power disconnect switches located in the main control room. This limits the effects of local fires by preventing propagation of erroneous commands to system electronics.

Fire protection design bases and requirements are documented in CESSAR-DC. Section 9.5.1 provides general fire protection requirements and Section 7.7.1.3.1 provides requirements specific to control panels. They will be implemented through procurement specifications as part of the process for procuring a owner/operators' plant. Independent review will verify that design requirements are implemented correctly in the specifications. Further assurance of compliance is provided by the design commitment in the Fire Protection System ITAAC requiring a fire hazards analysis, which considers the potential for fire hazards.

CESSAR DESIGN CERTIFICATION

The Nuplex 80+ main control panels are designed to ensure an adequate man-machine interface while meeting requirements for independence of redundant circuits. This is accomplished through a defense in depth approach that takes advantage of the intrinsic reliability of low energy circuits and the independence of the Main Control Panels and Remote Shutdown Panels. To minimize the potential for multiple channel damage within the Main Control Panels or Remote Shutdown Panels the following design features are employed:

A. Low energy circuits (less than 50 volts) are used to the maximum extent practical. This includes, for example, switch sense, lamps, indicators and alarm tiles.

B. Fire retardant (e.g., meeting UL-94 rating) materials are used throughout the panel enclosures, and the enclosures are equipped with smoke detectors. Fire resistant insulation material for control panel wiring meets the applicable requirements of IEEE Standard 383.

C. Electrical independence of channelized circuits is maintained throughout the panel enclosures.

Although the design features above minimize the potential for multiple redundant channel damage, the following design features accommodate such a catastrophic event:

A. All main control room circuits are fault isolated from the electronics to which they interface. Similarly, all remote shutdown panel circuits are fault isolated from the electronics. Therefore, the main control room and remote shutdown panel circuits are inherently isolated from each other and share no common failure modes.

B. All Main Control Panel and Remote Shutdown Panel circuits are passive. Momentary contacts are used for all switches with the memory of control panel commands retained only in electronics located in the I&C equipment rooms. This passive design is used for discrete state component controls as well as setpoint change commands and position change commands from process controllers for analog components. This passive design ensures that transfer of control from the main control room to the remote shutdown panel (or vice versa) is bumpless (i.e., no setpoints or component states will be affected). This design also ensures that all open circuit failures have no impact on control setpoints, modes or component states.

C. The main control room, remote shutdown panel and the I&C equipment rooms are each located in separate fire zones. Therefore, the plant can be safely shut down with a

Question 940308-13:

The single remote shutdown room (RSR) controls both Division I and II equipment. Is there a single fire location which could simultaneously impact the control cables (both Divisions) which link the control room with the RSR, thereby preventing the control of safe shutdown equipment from either the remote shutdown room or main control room? Please describe the plant arrangements and/or design requirements which provide confidence that there is not such a single fire location.

Response:

Following a fire, cold shutdown can be accomplished by using one of the two safety-related divisions from either the control room or the remote shutdown room. There is not a single fire location which can disable both divisions. This is accomplished by the following arrangement and design features:

Outside of containment, a 3-hour fire rated barrier is provided to separate the two divisions. Additional 3-hour fire rated barriers are also provided between quadrants within a division to meet regulations and to provide additional defense-in-depth.

The control room and remote shutdown room are physically and electrically isolated from each other. The control room is separated from the remote shutdown room with 3-hour fire rated barriers. There are four cable chases (one for each electrical channel) that enter the control room. Three-hour fire rated barriers separate each chase from the other. Each safety-related electrical channel enters the control room through its associated cable chase. The four safety-related electrical channels are also separated by 3-hour fire rated barriers before entering the remote shutdown room. The remote shutdown room is located on the Division I side of the divisional wall adjacent the Channel A and C cable chase on elevation 70+0. The A and C control channels enter the remote shutdown room from these chases through separate conduits located in the wall. The Division II, Channels B and D are routed from the Division II side to the remote shutdown room through separate conduits embedded in the floor. Electrical isolation from the control room is maintained by the remote shutdown room transfer switches so that a fire in the remote shutdown room cannot impact the control room.

Inside containment and annulus, cables required for safe shutdown are mineral insulated and 3-hour fire rated. The redundant shutdown paths are separated by either reinforced concrete walls, a component such as a steam generator or pressurizer, or by at least 20 feet with no intervening combustibles. A fire hazards analysis will confirm the likelihood that one of the two divisions are available following a fire inside containment.

Question 940308-14:

The Certified Design Material section on site envelope identifies a maximum precipitation for roof design. Is there a reason that it is limited to roof design? Are there other plant locations that are designed for less or more maximum precipitation rates?

Response:

The maximum precipitation is a site parameter and, therefore, structures and all site design are based on the same precipitation intensities given in the Certified Design Material Table 5.0-1 and CESSAR-DC Table 2.0-1. Certified Design Material Table 5.0-1 and CESSAR-DC Table 2.0-1 are being revised to remove "(for Roof Design)" and to clarify that the snow load is a design load.

TABLE 2.0-1

(Sheet 1 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Ground Water	
Maximum Level:	2 feet below grade
Flood (or Tsunami) Level[1]	
Maximum Level:	1 foot below grade
Precipitation (for Roof Design)	
Maximum rainfall rate:	19.4 in/hr. and 6.2 in/5 min.[2]
Maximum snow DESIGN load:	50 lb/sq. ft.
Design Temperatures	
Ambient	
1% Exceedance Values	
Maximum:	100°F dry bulb/77°F coincident wet bulb
	80°F wet bulb (non-coincident)
Minimum:	-10°F
0% Exceedance Values (Historical Limit excluding peaks < 2 hours)	
Maximum:	115°F dry bulb/80°F coincident wet bulb
	81°F wet bulb (non-coincident)
Minimum:	-40°F
Station Service Water Inlet:	95°F[3]
Condenser Circulating Water Inlet:	≤100°F

TABLE 5.0-1

SITE PARAMETERS

Maximum Ground Water Level	2 feet below finished plant grade level
Maximum Flood (or Tsunami) Level	1 foot below finished plant grade level
Precipitation (for Roof Design)	
Probable Maximum Precipatation (PMP) Estimate (Maximum Average Value Over One Square Mile Area for one hour)	19.4 inches per hour with a ratio of 0.32 for 5 minute to 1 hour PMP estimate. (6.2 inches per 5 minutes)
Maximum Snow DESIGN Load	50 pounds per square foot
Design Ambient Temperatures	
0% Exceedance Values (Historical Limit Excluding Peaks < 2 hours)	
Maximum	115°F dry bulb 80°F coincident wet bulb temperature
	81°F wet bulb (non-coincident) temperature
Minimum	-40°F
Extreme Wind	
Basic Wind Speed	110 miles per hour (50 year recurrence)
	122 miles per hour (100 year recurrence)
Tornado	
Maximum Tornado Wind Speed	330 miles per hour
Maximum Pressure Differential	2.4 pounds per square inch

Question 940308-15:

In the staff's FSER section regarding the design tornado windspeed, a reference is made to ABB-CE meeting the interim staff position in Reg. Guide 1.76, but it is not clear that is Reg. Guide 1.76 as modified by the letter of March 25, 1988. Is this made clear in the CE-SSAR?

FOR THE STAFF: Is there any ambiguity in the above FSER statement regarding the design tornado windspeed?

Response:

CESSAR-DC Table 2.0-1 Note 6, CESSAR-DC Section 3.3.2.1 and Reference 4 to "References for Section 3.3" are definitive on this issue.

The related CESSAR-DC sections are attached for information.

TABLE 2.0-1 (Cont'd)

(Sheet 2 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Extreme Wind	
Basic Wind Speed:	110 mph
Importance Factors:	1.0[4]/1.11[5]
Tornado[6]	
Maximum tornado wind speed:	330 mph
Rotational Speed:	260 mph
Translational velocity:	70 mph
Radius:	150 ft
Maximum pressure differential:	2.4 psi
Rate of pressure drop:	1.7 psi/sec
Missile spectra:	per SRP 3.5.1.4 Spectrum II
Soil Properties	
Minimum Bearing Capacity (demand):	12 ksf (static)[9]
Best Estimate of Minimum Shear Wave Velocity:	700 ft/sec[7]
Best Estimate of Liquefaction Potential:	None (at site-specific SSE level)
Seismology	
SSE Peak Ground Acceleration (PGA):	0.30 g [8]
SSE Response Spectra:	Section 3.7.1
SSE Time History:	Section 3.7.1
Aircraft Hazards	
Plant to airport distance	5mi.< D <10mi. with annual operation less than $5000D^2$ or D>10mi. with an annual operation less than $10000D^2$ (D = distance in miles)
Plant to edge of military training routes	D>5mi. with an annual operation less than 1000 flights (D = distance in miles)
Plant to edge of Federal airway, holding pattern, or airport	D>2mi. (D = distance in miles)

TABLE 2.0-1 (Cont'd)

(Sheet 3 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Meteorology

Short-term dilution factor	X/Q 1.0×10^{-3}; EAB = 500 meters
Long-term dilution factor	X/Q 2.2×10^{-5}; LPZ = 3000 meters

NOTES:

1. Probable maximum flood level (PMF), as defined in ANSI/ANS-2.8, "Determining Design Basis Flooding at Power Reactor Sites."

2. Maximum value for 1 hour 1 sq. mile PMP with ratio of 5 minutes to 1 hour PMP of .32, as found in National Weather Service Publication HMR No. 52.

3. Maximum normal power and normal shutdown temperature of the Station Service Water System Intake based on one percent exceedance meteorologic conditions. See item C of Section 9.2.5.1.3 for Ultimate Heat Sink temperature interface requirement for a design basis accident concurrent with a loss-of-offsite power.

4. 50-year recurrence interval; value to be utilized for design of non-safety-related structures only.

5. 100-year recurrence interval; value to be utilized for design of safety-related structures only.

6. 10,000,000-year tornado recurrence interval, with associated parameters based on the NRC's interim position on Regulatory Guide 1.76. Pressure effects associated with potential offsite explosions are assumed to be non-controlling for the design.

7. Site profiles are given in Section 2.5. Profiles include consideration of variability of soil properties. The lower bound of best estimate of soil shear wave velocity defines the lower bound of dynamic Soil-Structure Interaction analysis of the superstructure.

8. The control motions are defined in Section 2.5.

9. Bearing capacity is defined at the foundation level of the Nuclear Island structure.

3.3 WIND AND TORNADO LOADINGS

All Seismic Category I structures, except those not exposed to wind, are designed for wind and tornado loadings.

3.3.1 WIND LOADINGS

The design for wind loading is in accordance with ANSI/ASCE 7, "Minimum Design Loads for Buildings and Other Structures" (Reference 1). Structural geometries not addressed in ANSI/ASCE 7 shall be evaluated using ASCE Paper 3269, "Wind Forces on Structures" (Reference 2), and ASCE Paper 4933, "Wind Loads on Dome-Cylinder and Dome-Cone Shapes" (Reference 3).

3.3.1.1 Design Wind Velocity

A design wind velocity of 110 mph, at a height of 33 feet above nominal ground elevation is used as the maximum wind speed for a 50 year recurrence period.

Velocity profiles and associated effective pressures for winds with a 100 year recurrence period are calculated in accordance with Section 6 of Reference 1 utilizing an Importance Factor, I, of 1.11 and Exposure C.

Gust response factors are dependent on height above grade level and are in accordance with Table 8 of Reference 1 for Exposure C.

3.3.1.2 Determination of Applied Forces

Based on structure geometry and physical configuration, the effective pressure distribution is transformed into applied equivalent static building forces utilizing appropriate shape coefficients given in Reference 3.

Wind pressure distribution curves for the containment shield building are shown in Figure 3.3-1. The maximum height of the shield building above grade is approximately 173 feet 3 inches.

3.3.2 TORNADO LOADINGS

All Seismic Category I structures that perform a safe shutdown or accident mitigation function, except those structures not exposed to wind, are designed for tornado loadings.

3.3.2.1 Applicable Design Parameters

Tornado effects are in accordance with Interim Regulatory Guide 1.76 (Reference 4). The following parameters are applicable to the design basis tornado.

Maximum wind speed:	330 mph
Rotational speed:	260 mph
Translational velocity:	70 mph
Radius:	150 feet
Maximum pressure differential:	2.4 psid
Rate of pressure drop:	1.7 psi/second
Missile Spectra:	See Table 3.5-2

3.3.2.2 Determination of Forces on Structures

The forces on Seismic Category I structures due to tornado wind loadings are obtained using methods outlined in Section 3.3.1.2, with a wind velocity of 330 mph (vector sum of all component velocities - assumed constant with height). Velocity profiles are determined as outlined in Section 3.3.1.1. Effective pressure distribution loads are transformed into equivalent static building forces as outlined in Section 3.3.1.2. In determining tornado wind loadings, both the importance factor and gust factors are taken as unity.

Tornado loadings include tornado wind pressure, internal pressure due to tornado created atmospheric pressure drop, and forces generated due to the impact of credible tornado missiles. These loadings are combined with other loads as described in Section 3.8.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

Adjacent structures will not be permitted to affect or degrade the capability of Seismic Category I structures to perform their intended safety functions as a result of tornado loadings. This is accomplished by one of the following methods:

A. Designing the adjacent structure to Seismic Category I tornado loadings.

B. Investigating the effect of adjacent structural failure on Seismic Category I structures to determine that no impairment of function results.

C. Designing a structural barrier to protect Seismic Category I structures from adjacent structural failure.

REFERENCES FOR SECTION 3.3

1. "Minimum Design Loads for Buildings and Other Structures," ANSI/ASCE 7.

2. "Wind Forces on Structures," ASCE Paper No. 3269, Transactions, ASCE, Vol. 126, Part II, 1961, p. 1124.

3. "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," ASCE Paper No. 4933, Journal of the Structural Division - Proceedings of the American Society of Civil Engineers, Vol. 92, No. ST5, October 1966.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation of Recommended Modification to the R.G. 1.76 Tornado Design Basis for the ALWR, attached to a March 25, 1988 NRC letter to the ALWR Utility Steering Committee.

Question 940308-16:

You have stated that the sources of missiles will be minimized by equipment design features. Where are these specific design features specified and to what extent do they include the design of non-seismic structures such as turbine building siding which could be blown off in a tornado?

Response:

The design features which reduce the potential of missiles are identified in CESSAR-DC Section 3.5. Exterior safety related equipment is protected by structures or is hardened to withstand the externally generated tornado missile spectra identified in Section 3.5. The spectra are from NUREG-0800, NRC Standard Review Plan. The spectra do not specifically identify siding blown off the turbine building. The tornado missile spectra identified envelope credible tornado missiles. CESSAR-DC Section 3.5 is being revised from "Minimizing the sources of missiles..." to "Reducing the potential for sources of missiles...".

A marked CESSAR-DC revision for Amendment V is attached.

3.5 MISSILE PROTECTION

The missile protection design for Seismic Category I structures, systems and components is described in this section.

Missile protection or redundancy is provided for Seismic Category I equipment and components such that internal and external missiles will not cause the release of significant amounts of radioactivity or prevent the safe and orderly shutdown of the reactor.

The protection of essential structures, systems and components will be accomplished by one or more of the following: D

A. Reducing the potential for ~~Minimizing the~~ sources of missiles by equipment design features that prevent missile generation.

B. Orientation or physical separation of potential missile sources away from safety-related equipment and components.

C. Containment of potential missiles through the use of protective shields and barriers near the source.

D. ~~The~~ Hardening of safety-related equipment and components to withstand missile impact, where such impacts cannot be reasonably avoided by the methods above. I

3.5.1 MISSILE SELECTION AND DESCRIPTION

Potential missiles are identified and characterized by type and source and their probability of occurrence, retention and impact. For equipment with energy sources capable of creating a missile, the selection is based on the application of a single-failure criterion to the retention features of the component. Where sufficient retention redundancy is provided in the event of a failure, no missile is postulated. D

Internally generated missiles can be generated potentially from two types of equipment: rotating components and pressurized components. Rotating components include turbine wheels, fans, auxiliary pumps and their associated motors. Pressurized components include valves, heat exchangers, vessels and their associated components.

Question 940308-17:

Please describe the analysis which showed a maximum 2 psi vacuum in containment following inadvertent containment spray actuation. Include initial conditions and assumptions. If initial temperature is 110 F, should there be a Tech Spec limit for minimum containment temperature?

Response:

The reduction in containment pressure following an inadvertent containment spray actuation was calculated in the following way. The containment spray was assumed to cool the containment atmosphere from its initial temperature to the minimum temperature of the spray water entering the containment. The decrease in containment pressure results from the changes in the air and steam partial pressures. The relative humidity at the beginning and final condition was assumed to be 100%.

It is conservative to assume maximum containment temperature and minimum spray water temperature. It is also conservative to assume the minimum initial containment pressure. The relationship between the minimum allowable IRWST water temperature and the containment temperature is given in Technical Specification Figure 3.5.4-1. The maximum containment atmosphere temperature is limited to 110°F in the Technical Specification. The minimum containment atmosphere pressure is limited to 14.3 psia in the Technical Specifications.

With a containment atmosphere temperature of 110°F and a spray water temperature of 81°F, the resulting pressure differential across the containment steel shell is 1.82 psig if the initial containment atmosphere pressure is greater than 14.3 psia.

Question 940308-18:

Does the CE-SSAR require that all valves be designed with a shoulder or backseat larger than the bonnet opening such that stem ejection is prevented, consistent with the staff's FSER statement on page 3-39? Could valve bonnets or bonnet bolts be considered potential missiles? If so, is this considered in the CE-SSAR?

Response:

CESSAR-DC does not state that there are no missiles from valve stems. CESSAR-DC, Section 3.5 states that "there are no missiles postulated from valves..." for reasons given therein. Justification provided addresses valve stems, bonnets, operators and diaphragms.

Except for action relating to the turbine maintenance and inspection program, CESSAR-DC does not currently include a COL action item requiring confirmation of missile protection assumptions. A COL action item is being incorporated into CESSAR-DC Section 3.5.4 in Amendment V.

Related CESSAR-DC sections including a marked revision for Amendment V are attached for information.

The types of missiles considered and/or not considered in the design of Seismic Category I structures, systems, and components are discussed in the following sections:

A. Internally Generated Missiles (Outside Containment), described in Section 3.5.1.1.

B. Containment Internal Missiles, defined in Table 3.5-1 and Section 3.5.1.2.

C. Turbine Missiles, described in Section 3.5.1.3.

D. Natural Phenomena (Tornado) Missiles, described in Section 3.5.1.4.

E. Site Proximity Missiles (Except Aircraft), described in Section 3.5.1.5.

F. Aircraft Hazards, described in Section 3.5.1.6.

D

3.5.1.1 Internally Generated Missiles (Outside Containment)

Internally generated missiles (outside containment) from rotating and pressurized components are not considered credible for the reasons discussed below.

The redundant safety systems outside of containment are physically separated such that no single gravitational or other type missile can impact both systems. I

3.5.1.1.1 Auxiliary Pumps and Motors

D

There are no postulated missiles originating from auxiliary pumps and associated motors outside containment for the following reasons:

A. The pump motors are induction type which have relatively slow running speeds and are not prone to overspeed. I The motors are all pretested at full running speed by the motor vendor prior to installation.

D

B. In addition to the low likelihood of missiles due to motor overspeed as discussed in A. above, the motor stator would tend to serve as a natural container of rotor missiles if there were to be any.

C. All pumps normally have relatively low suction pressures I and, therefore, would not tend to be driven to overspeed due to a pipe break in the discharge line. In addition, the induction motor would tend to act as a brake to prevent pump overspeed.

D. Industry pump designs are such that (and service history shows) no occurrences of impeller pieces penetrating pump casings.

3.5.1.1.2 **Emergency Feedwater Pump Turbines**

There are no postulated missiles from the Emergency Feedwater (EFW) pump turbines for the following reasons:

A. Turbine overspeed protection; electrical trip at 115% of rated speed, and mechanical trip at 125% of rated speed.

B. Assurance of turbine disk integrity by design and inspection.

C. Enclosure of the EFW pumps and turbine drivers in a reinforced concrete room.

3.5.1.1.3 **Valves**

There are no missiles postulated from valves for the following reasons:

A. All valve stems are provided with a backseat or shoulder larger than the valve bonnet opening.

B. Motor operated and manual valve stems are restrained by stem threads.

C. Operators on motor, hydraulic and pneumatic operated valves prevent stem ejection.

D. Pneumatic operated diaphragms and safety valve stems are restrained by spring force.

E. All valve bonnets are either pressure sealed, threaded or bolted such that there is redundant retention for prevention of missile generation.

3.5.1.1.4 **Pressure Vessels**

All pressurized vessels outside containment are moderate energy (275 psig) or less and are designed and constructed to the standards of the ASME Code. In addition to the ASME Code examination and testing requirements, all vessels will receive periodic in-service inspections. Where appropriate, these components are provided with pressure relief devices to ensure that no pressure buildup will exceed material design limits.

On this basis, moderate energy pressure vessels are not considered credible missile sources.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Table 3.5-1 lists postulated missiles from equipment inside containment, and summarizes their characteristics. Included are major pretensioned studs and nuts, instruments, and the CEDM missile. Other items which were considered and specifically excluded because of redundant retention features are valve stems, valve bonnets and pressurized cover plates.

3.5.1.3 Turbine Missiles

The probability of turbine missile generation and adverse impact effects on Seismic Category I systems and components is assured to be less than 1.0E-4 events per turbine-year by a combination of the following measures:

A. Reliable turbine overspeed protection provisions (see Section 10.2.2 for details).

B. Adequate assurance of turbine disc integrity by design and inspection (see Sections 10.2.2 and 10.2.4 for details).

C. Placement and orientation of the turbine generator (described below).

D. The protection provided by plant structures, not explicitly designed as barriers, that may reduce missile energy to less than that required to penetrate Seismic Category I structures.

E. Adequate turbine maintenance and inspection program (see Section 10.2.4 for details).

The turbine generator placement and orientation for the System 80+ Standard Design, and the corresponding low-trajectory missile strike zones, are illustrated in Figure 1.2-1. The placement and orientation of the turbine generator provides adequate protection against low trajectory turbine missiles by excluding safety-related structures, systems, and components from the low trajectory turbine missile strike zones in accordance with the guidelines of Regulatory Guide 1.115.

Critical structures (i.e., those housing safety-related equipment) and exterior equipment are located in line with, or within close proximity to, the longitudinal axis of the turbines. This makes the potential for turbine-generated missiles to strike these targets negligibly small.

3.5.4 GENERAL DESIGN BASES

credible

Protection for all Seismic Category I structures, systems and components are provided by the following:

A. For systems and parts of systems located inside the containment (RCS and connected systems, Engineered Safety Feature systems), appropriate missile barrier design procedures are used to ensure that the impact of any potential missiles will not lead to a loss-of-coolant-accident or preclude the systems from carrying out their specified safety functions.

B. For systems and equipment outside containment, appropriate design procedures (e.g., proper turbine orientation, natural separation, or missile barriers) are used to ensure that the impact of any potential missiles does not prevent the system or equipment from carrying out its specified safety function.

credible

C. For all systems and equipment, appropriate design procedures are used to ensure that the impact of any potential missiles does not prevent the conduct of a safe plant shutdown, or prevent the plant from remaining in a safe shutdown condition.

D. Safety-related instrumentation and control equipment are protected from potential missile sources. The 1E and associated cabling and sensing lines are also protected from potential missile sources.

The COL applicant will ensure that as-built conditions provide Seismic Category I structures, systems and components protection from credible potential missiles.

Question 940308-19:

Please clarify the depth of the CTG building foundation and the height of this building above ground, and the elevation and seismic qualification of the CTG fuel tanks.

Response:

The Combustion Turbine Generator (CTG) Alternate AC source is described in CESSAR-DC Sections 1.2.16.2 and 8.3.1.1.1.5. The CTG is founded on a concrete reinforced pad at plant finished grade. The CTG is provided as a packaged unit and is housed in its own metal enclosure/support structure. The CTG fuel storage tanks are steel tanks constructed on reinforced concrete foundations. The CTG and its foundation will be designed such that the maximum possible flood elevation will be a minimum of one foot below the lowest CTG operating component.

Question 940308-20:

FOR THE STAFF: On page 3-41 the staff states that the orientation of the turbine generator system is such that any potential turbine missiles will not strike the reactor building. It appears that this staff position declares that NO possible turbine missile will strike the reactor building. If so, how is this justified?

Question 940308-21:

FOR THE STAFF: Please clarify what is meant by "continuously manned" on FSER page 19-220. Where and by whom?

Question 940308-22:

Please compare the elements of your shutdown risk PRA (i.e. loss od DHR, LOCA, LOOP, and fire) with those from the NSAC 84 and Seabrook studies.

Response:

The following table compares the Core Damage Frequency (CDF) for the ABB-CE, NSAC-84 and Seabrook studies. All three studies concluded that human actions are important in shutdown modes. The NSAC-84 and Seabrook studies concluded that the uncertainties were greater than for power operation and the ABB-CE analysis did not address uncertainty.

COMPARISON OF SHUTDOWN PRAs

Event Type	ABB-CE	NSAC-84	Seabrook
Total CDF	8.4×10^{-7}	1.8×10^{-5}	4.5×10^{-5}
Contribution for reduced inventory	31%	61%	71%
Contribution by initiating event			
Loss of DHR	23%	74.6%	61%
LOCA	16%	14.7%	18%
LOOP	25%	0.7%	6%
Fire	36%	Not Analyzed	4%
Seismic or Loss of Support Systems	Not Analyzed	Not Analyzed	11%
Sequences not specified	0	10%	0

This information supplements the information presented in CESSAR-DC Table 19.8.1-3.

Question 940308-23:

Please provide the code manual which deals with the models, correlations, and basis for the ATHOS code.

FOR THE STAFF: How did you address the issue of fluid elastic instabilities in the steam generator secondary side (generating cross-flow conditions across the tubes) which may lead to long term tube vibration and resultant cracking and tube rupture?

Response:

For the purpose of expediency, the requested ATHOS code manual (EPRI Report NP-2698-CCM, dated October 1982) has been sent directly to Dr. Catton (copy of manual cover page attached).

Further, in response to a steam generator question asked at the February 9, 1994 Subcommittee meeting (Question No. 940209-2), ABB-CE indicated that a white paper was being prepared regarding System 80+ steam generator design issues and that a copy of the paper would be provided to the ACRS, if desired. Internal review of this paper is expected to require another 4-6 weeks. In lieu of the paper, a presentation will be made at the April 5-6, 1994 ACRS Subcommittee meeting which addresses the significant issues discussed in the paper and which also provides the pertinent design, analysis, and operational data contained in the white paper.

Some of the information to be presented (3 pages) is proprietary in nature. ABB-CE has already sent a letter to NRC staff (LD-94-020, dated March 25 , 1994) to document the proprietary classification for this information so that it can be presented at the Subcommittee meeting.

ATHOS—A Computer Program for Thermal-Hydraulic Analysis of Steam Generators
Volume 1: Mathematical and Physical Models and Method of Solution

NP-2698-CCM, Volume 1
Research Project 1066-1

Computer Code Manual, October 1982

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Volume 2: Programmer's Manual

NP-2698-CCM, Volume 2
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Analysis and Testing Program
Nuclear Power Division

ATHOS—A Computer Program for Thermal-Hydraulic Analysis of Steam Generators
Volume 3: User's Manual

NP-2698-CCM, Volume 3
Research Project 1066-1

Computer Code Manual, October 1982

Prepared by

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Question 940308-24:

On CE-SSAR pages 19.7-31 and 19.7-32, please provide the basis for why the fire barrier failure probability (1.2E-3) is so low?

Response:

The fire barrier failure rate of $1.2x10^{-3}$ per demand that was used in the System 80+ quantitative fire scoping analysis (CESSAR-DC page 19.7-31) was selected from a set of values presented in Section 4.6 of NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150", dated November, 1990. This failure rate was primarily used to represent the failure of the reinforced concrete wall between the diesel generator rooms and the adjacent areas in tne subsphere at elevation 50+0. It should be noted that at least one additional fire barrier would have to fail for the fire to propagate to another fire area containing safety-related equipment.

Enclosure II to LD-94-023

ABB Combustion Engineering

System 80+™ Standard Plant Design

Revised Responses to ACRS ABB-CE Standard Plant Designs Subcommittee Questions from Previous Meetings

Revised Responses to ACRS ABB-CE Standard Plant Designs Subcommittee Questions from Previous Meetings

Question 931208-10:

What is the basis for the test requirement specified in ITAAC test #9.b) on Table 2.6.1-1?

Revised Response:

It was determined that the basis for ITAAC test #9.b) is the Electrical Power Distribution System's capability to operate the Class 1E leads at the loads' analyzed minimum voltage. ITAAC #9.b) Inspections, Tests, Analysis, and Acceptance Criteria will be modified as follows to reflect this basis. The modified item #9.b) will also be relocated to Item #22 (appearing as Item #22.b)), since Item 22 deals specifically with voltage analysis.

Inspections, Tests, Analyses	Acceptance Criteria
22.b) Tests of the as built Class 1E EPDS will be performed by operating connected Class 1E loads at the analyzed minimum voltage.	22.b) Connected Class 1E loads operate at the analyzed minimum voltage as determined by the voltage drop analysis.

Question 931208-11:

For those plant components identified in the SSAR or Design Description which are not expected to be replaced during a 60 year design life, how will it be assured that such components will be designed to last 60 years? Will ABB-CE provide any design guidance for when to replace components that are not expected to last 60 years?

For NRC STAFF: What is the staff's position on the degree to which ABB-CE needs to specify design guidance in both of the above cases?

Revised Response:

The 60-year plant design life for System 80+ is accomplished through a combination of design, material selection, design analysis, procurement requirements, pre-/inservice inspection, maintenance, and replacement activities for individual components and systems. These issues are addressed in CESSAR-DC. Major NSSS component design life (e.g., the reactor vessel) is confirmed through analysis and monitored throughout the station life. For example, CESSAR-DC Section 3.9.1.1 describes the approach to fatigue analysis of the NSSS based on the transients expected through a 60-year plant life. For the reactor vessel, improved material specifications are stated in Section 5.2.3.2 and the neutron irradiation analysis for 60 years is documented in Section 5.3.2.1. Major Nuclear Steam Supply System (NSSS) and turbine generator (T-G) components are not expected to be replaced during the 60-year design life of the System 80+ plant. The design and inspection requirements for the turbine are reviewed in CESSAR-DC Sections 10.2.3 and 10.2.5.

For components that may be replaced during the 60-year design life, the design life is determined during the actual design and procurement process where the objective is to obtain the longest design life obtainable for a given component, consistent with life-cycle cost considerations. For those components where 60-year life is not obtained, replacement criteria will be included in the plant's equipment manuals and, as part of maintenance and surveillance programs as described in Section 3.11.2.1 and summarized in Section 3.9.6 and 6.6.

Further, the ALWR Utility Requirements Document (URD), Volume II, Chapter 1, Section 11.3.4.1, requires that the design life plan and preventive maintenance program include monitoring of all structures, systems, and components (SSCs) for which monitoring is cost effective with respect to safety, reliability, and maintenance.

Compliance with the URD is a "living" process in that compliance with major URD requirements is established early in the design process, and compliance with other more-detailed URD requirements is documented as the design is completed and corresponding design detail becomes available. For example, the Nuplex 80+ Advanced Control Complex design includes monitoring and recording capability for all significant components and parameters, which would support any practical design life or maintenance plan. Detailed design life planning and maintenance programs are developed during the final stages of plant design when detailed design information is available, including specific equipment to be

Revised Response (continued):

procured. Proposals to prospective customers include programs to address periodic monitoring, continuous monitoring, data management systems, monitoring techniques and frequencies, and training.

Moreover, the System 80+ Design Reliability Assurance Program (D-RAP) provides input to the plant's Operations Reliability Assurance Program (O-RAP), as described in Section 17.3 of CESSAR-DC. To ensure that the owner/operator reviews the guidance in the URD at the time the O-RAP is developed, a statement referencing the URD is being added to Section 17.3.9 of CESSAR-DC, as shown on the attached mark-up.

Section 11.3.4.1) for guidance on monitoring and maintenance programs.

availability targets and the associated maintenance practices for achieving them.

17.3.8 DEFINING FAILURE MODES

The determination of dominant failure modes of risk-significant SSCs will include historical information, analytical models and existing requirements. Many PWR systems and components have compiled a significant historical record, so an evaluation of that record comprises Assessment Path A in Figure 17.3-3. Details of Path A are shown in Figure 17.3-4.

For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary, shown as Assessment Path B in Figure 17.3-3. The details of Path B are given in Figure 17.3-5. The failure modes identified in Paths A and B are then reviewed, including the existing maintenance activities in the industry and the maintenance requirements (Assessment Path C in Figure 17.3-3). Detailed steps in Path C are outlined in Figure 17.3-6.

17.3.9 OPERATIONS RELIABILITY ASSURANCE ACTIVITIES

Once the dominant failure modes are determined for risk-significant SSCs, an assessment should be used to determine suggested O-RAP activities that will assure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, and/or periodic preventive maintenance (Reference 17.3-1). An example of a decision tree that would be applicable to these activities is shown in Figure 17.3-7. As indicated, some SSCs may require a combination of activities to assure that their performance is consistent with the PRA. In addition, the owner/operator should review the ALWR Utility Requirements Document (Volume II, Chapter 1,

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to assure that they will respond to appropriate signals, and inspection of passive SSCs (such as tanks and pipes) to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurement of output (such as pump flow rate or heat exchanger temperatures), measurement of magnitude of an important variable (such as vibration or temperature), and testing for abnormal conditions (such as oil degradation or local hot spots).

Periodic preventive maintenance is an activity performed at regular intervals to preclude problems that could occur before the next preventative maintenance (PM) interval. This could be regular oil changes, replacement of seals and gaskets, or refurbishment of equipment subject to wear or age-related degradation. The designer could provide the COL applicant with recommended reliability activities such as providing limitations for assuring reliability, and methods to determine service life, if known.

ATTACHMENT 5

NRC Staff Responses to ACRS Questions on the System 80+ Standard Plant Design

(March 8 and 9, 1994 Questions)

940308-~~18~~ 21 Carroll FOR THE STAFF: Please clarify what is meant by "continuously manned" on FSER page 19-220. Where and by whom?

The complete sentence reads as follows: "ABB-CE stated that the flood door open/close status will be continuously monitored and manned 24 hours a day." The intent of this sentence is to state that the open/close status of the door in question will be continuously monitored and the monitoring station will be continuously manned. The monitoring station is located in the security Central Alarm Station (CAS). The CAS is continuously manned and monitored by security personnel. ABB-CE indicated to the staff that if an alarm indicates that a door is mispositioned, a security officer is sent out to investigate. Security procedures are the responsibility of the COL applicant as per Chapter 13 of the CESSAR-DC. Also, communications are provided between the main control room and the security CAS. Fire alarm information is also retrievable in the main control room.

In the Design Description of ITAAC 2.1.1 it is stated that " Flood doors, shown on Figures 2.1.1-1 through 2.1.1-12, have sensors with open and closed status displays provided at a central fire alarm station." This issue was also address by ABB-CE as an Action Item on Shutdown Risk in an NRC meeting July 7-8, 1993. The statement in the FSER is based on the ITAAC information and the written response provided to the staff by ABB-CE as a result of the meeting.

The staff is reverifying this information with ABB-CE. The staff will clarify the FSER to reflect the above information.

940308-15 Lindblad In the staff's FSER section regarding the design tornado windspeed, a reference is made to ABB-CE meeting the interim staff position in Reg. Guide 1.76, but it is not clear that is Reg. Guide 1.76 as modified by the letter of March 25, 1988. Is this made clear in the CE-SSAR?

FOR THE STAFF: Is there any ambiguity in the above FSER statement regarding the design tornado windspeed?

Response:

In FSER Section 3.3.2, the staff states that the staff's interim position on RG 1.76 was issued in a March 25, 1988 letter. It later states that all seismic Category I structures, exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist tornado effects in accordance with the interim staff position in RG 1.76. The FSER does not appear to be ambiguous on this point.

However, the introductory sentence seems to be misleading in that it states that the current staff position with regard to design-basis tornados is contained in two documents written in 1974 (WASH-1300 and RG 1.76). This sentence will be changed to state, "The staff position with regard to design-basis tornados was previously based on two documents written in 1974..."

940308-20 Carroll FOR THE STAFF: On page 3-41, the staff states that the orientation of the turbine generator system is such that any potential turbine missiles will not strike the reactor building. It appears that this staff position declares that NO possible turbine missile will strike the reactor building. If so, how is this justified?

Response:

The sentence in the FSER overstated its case and will be revised to state, "With respect to the reactor building, the turbine system is oriented so that a postulated turbine missile is not likely to strike the reactor building." This statement is more consistent with the remainder of the FSER section which discusses the probability of adverse effects due to turbine missiles.

940308-23 Catton Please provide the code manual which deals with the models, correlations, and basis for the ATHOS code.

FOR THE STAFF: How did you address the issue of fluid elastic instabilities in the steam generator secondary side (generating cross-flow conditions across the tubes) which may lead to long term tube vibration and resultant cracking and tube rupture?

Response:

Steam generator tube integrity and the potential for containment bypass from a steam generator tube rupture are currently discussed in FSER Sections 5.4.2 and 15.3.9 respectively. This issue was identified as an open item in the DSER under DSER Open Item 15.3.8-1. The staff adopted a defense-in-depth approach to this SECY-93-087 issue in the context of accident prevention and mitigation. The first aspect involved the preventive measures that ABB-CE used to reduce the likelihood of a steam generator tube rupture (SGTR). The second aspect involved the mitigative measures that the System 80+ design contains that reduce the likelihood of a containment bypass sequence given an SGTR event. The principal concern for this issue involves the potential for a main steam safety valve (MSSV) sticking open during an SGTR resulting in an unisolable release path from the reactor coolant system to the environment.

Throughout the course of the staff's review of the System 80+ design, the staff has emphasized the importance of both prevention and mitigation measures in dealing with steam generator tube ruptures in ALWRs. Discussions were held between the staff and ABB-CE to go over preventive improvements offered in the System 80+ SG design. These improvements were based on years of design evolution of ABB-CE designed steam generators. The attached pages contain an extract from a Palo Verde report (reference 1) that summarizes the design evolution of ABB-CE steam generator designs. The staff has followed these improvements through work with the industry and has reviewed this experience for applicability to the System 80+ steam generators.

With respect to the ACRS's concern of fluid elastic instability, the vendor has used operating experience as-well-as new analytical tools to aid in the improvement of SG performance. This operating experience and analytical methods were discussed with the staff throughout the review process. As noted in the attached report, flow induced wear has been experienced in ABB-CE designed SGs. Two particular areas of concern exist (1) batwing (BW) wear, and (2) flow bypass resulting in high velocity fluid exiting above the economizer (System 80 design) causing corner tube wear.

For item (1) above (batwing wear), the System 80 design introduced an axial flow economizer section that changed the flow characteristics in the BW region. The upward fluid velocity in the central cavity increases with increasing height above the tube sheet

As part of the axial flow economizer design, the normally open tube lane and central cavity are filled with the economizer divider plate and support cylinder. Therefore the fluid in the central cavity does not begin its upward acceleration at the tubesheet but rather above the economizer. The resulting velocities at the batwing level are reduced as compared with non-economizer units that have the batwing feature. As a result of these design changes, BW wear is expected to be less severe (or potentially non-existent) in System 80 and System 80+ steam generators. Some tubes have been observed to have wear at the BW locations in the Palo Verde (System 80) steam generators. It should be noted that the BW feature was eliminated beginning with ABB-CE's Korean units and does not appear on the System 80+ steam generators.

For item (2) above (economizer corner region bypass flow induced wear), System 80+ contains a re-designed downcomer (cold side) recirculation window to eliminate the corner tube wear problem.

FSER Section 5.4.2 also contains a discussion of System 80+ SG design improvements with respect to hydraulic stability. The following discussion is from FSER Section 5.4.2:

> ABB-CE stated that the steam generator internal components are designed to maintain localized fluid velocities below the critical velocities which will cause excessive tube vibration. The horizontal egg crate tube supports are designed and located to maintain the natural frequency of the tubes higher than the exciting frequency induced by cross-flow in the tube bundle entrance region. In comparison to the System 80 steam generators, several design improvements, described below, have been made to the System 80+ steam generators to prevent the type of vibration induced wear seen at other plants:
>
> - The economizer divider plate and center stay [support] cylinder have been extended up to the top of the cold side recirculating fluid entrance window, to reduce fluid cross flow velocities.
> - The downcomer partition between hot side and cold side has been extended down to the top of the economizer divider plate to prevent circumferential flow.
> - The economizer flow shroud has been extended up 41 cm (16 in.) to meet the downcomer flow shroud, for a circumferential distance of 18 cm (7 in.) on each side of the cold side recirculating fluid entrance window, to limit the cross flow velocities in the open tube lane along the economizer divider plate.
> - One additional egg crate tube support has been added to the cold leg economizer region to stiffen the tubes. The combination of stiffer tubes and lower local fluid velocities results in a reduction in flow induced tube vibration for

the System 80+ steam generators.

A flow induced vibration test has been performed on a test model for the steam generator economizer and lower tube bundle region. This test model incorporated the design improvements described above for the System 80+ steam generators. ABB-CE states that the results of the economizer test should provide comparable results for the System 80+ units and, therefore, excessive tube vibration due to flow induced vibrations should not occur.

It should be noted that the Palo Verde SGTR was not the result of fluidelastic ~~flow~~ instability. The root cause has been attributed to outer diameter stress corrosion cracking (ODSCC). ODSCC for the Palo Verde Unit 2 SGs has been attributed to ridge deposits that formed in the free span (upper bundle) region due to water chemistry problems exacerbated by a resin release and SG thermal-hydraulic attributes that create the potential for dryout in the free span region.

ABB-CE also used the ATHOS-II code to aid in predicting potential thermal-hydraulic related problems for the System 80+ SGs. The results of the analysis showed thermal profiles in the free span region that have a lower steam quality and subsequently a reduced potential for dryout when compared to Palo Verde. This was due to the lower Thot (reduced from 621°F to 615°F) as-well-as an increased recirculation ratio from 3.00 (Palo Verde) to 4.05 (System 80+) for 100 percent power operation.

The staff discussed with ABB-CE their use of the ATHOS code as a design tool in System 80+ SG development; however, the staff did not rely on the ATHOS code for its safety conclusions nor was the code submitted for NRC review. ABB-CE also used the CEPAC code (described in Section 4 of CESSAR-DC Appendix 5F) to model plant response to an SGTR. The CEPAC code was not submitted for staff review; however, the code was benchmarked against other approved codes such as CENTS for the SGTR event.

Additionally, the staff focused the SGTR review on reducing the likelihood of a containment bypass sequence given the SGTR event. Consequently, the staff did not rely on preventive measures alone for a safety finding on risk associated with SGTRs/containment bypass. The staff pursued this review approach due to the history of steam generator problems experienced in the industry, the varying phenomena experienced in tube degradation mechanisms (e.g. vibration related failures, primary water SCC, ODSCC, dryout/steam blanketing), and the potential for an unidentified degradation phenomena to occur in a new SG design.

In response to the staff's approach, ABB-CE was charged with providing a detailed examination of the System 80+ plant behavior during an SGTR event. Also, ABB-CE was tasked with evaluating potential design alternatives that could reduce or eliminate the potential for MSSV lift. This detailed analysis is provided in CESSAR-DC Appendix 5F and contains both single and multiple (five) tube rupture analysis.

ABB-CE's design analysis and the staff's review resulted in several design modifications that would enhance the plant's and operator's response to a tube leak and/or a tube rupture event. The staff determined that with the (1) modification of the component coolant water system to ensure continued operation of the turbine bypass system throughout an SGTR event, (2) addition of two nitrogen-16 (N-16) monitors (one per SG) in the steamlines to provide early indication of tube leakage and to help SGTR diagnostics, (3) addition of associated ITAAC and TSs to ensure inclusion and availability of N-16 monitors, and (4) modification of emergency operations guidelines to ensure proper guidance of SGTR recovery actions, the System 80+ design provides adequate diagnostics and operator response time to mitigate the consequences of SGTR events (see FSER section 15.3.9 for additional information).

Reference 1: Palo Verde Unit 2 Steam Generator Tube Rupture Report, July 18, 1993; Docket No. STN 50-529.

Attachment

XIV. APPENDICES

A. Design Evolution of Combustion Engineering Steam Generators

The Design Evolution of the System 80 Steam Generator Support Structure was reviewed for structural adequacy. The Steam Generators (SG's) at PVNGS are of the System 80 design and were manufactured by Combustion Engineering (CE) at the Chattanooga, TN facility during the late 1970's. The design of the System 80 SG internal tube bundle support structure evolved from the earlier SG's that CE designed, manufactured and sold to the various electric power utilities during the 1960's and 1970's. Tube supports are required at periodic intervals along the U-tube to prevent flow induced vibration which can result in fretting-wear and/or fatigue failure. The changes to the design from the earlier Units were basically a result of trying to balance two opposing design parameters.

- The desire to maintain a large number of supports and their associated rigidity to provide the large margin (i.e. the "over-design" margin) for operating loads (mechanical stresses and flow induced vibration) and accident loads (i.e. LOCA, MSLB); verses
- The empirical evidence from the operating Units showed that the higher the number of supports, the more crevices are created in which corrosion products will accumulate, resulting in more plugged tubes.

The area in which there was the most "change" in the design of supports, in the history of the CE SG's, was in the upper bundle region; namely the partial eggcrates, the batwings (BW's) and the vertical support grids(VS's). The following outline details this design evolution.

1. "Early Units"

The "Early Units" consisted of Palisades, Mihama-1 (Westinghouse), and Fort Calhoun. The overall design of these SG's are too varied to be grouped into any specific category. However, they did have one common feature, the eggcrate design of their vertical support (VS) region was such that they used scallop bars to lock in the horizontal span of the tubes. (The VS region is basically an eggcrate support that lays on its side, with respect to the full and partial eggcrate supports below. However, to manufacture the same diamond pattern of the full and partial eggcrates in the VS region is unreasonable. Thus, the VS region uses a square eggcrate design with tubes in every other square. This increases tube spacing from 1" in the diamond pattern of the lower eggcrates to 1.25" in the VS region of the Early Units, which aided manufacturing [and aided Engineering by allowing the fluid to exit the bundle with less resistance]. However, the tubes in the VS region were still close enough that a

straight locking bar could not be used to lock the tube into its square eggcrate during assembly. A scalloped shaped locking bar was used).

Note: When comparing fossil boilers to nuclear SG's, the nuclear SG's were "unique" due to the boiling takes place on the shell side as opposed to the tube side. As a result, minimal flow induced vibration data was available to the designers of the CE SG's. To ensure the absence of flow induced vibration, the tube support structure in the early units were intentionally over-designed from a vibration standpoint. The opposing effect of this "over-designing" was that the supports were partially shielding the tubes, and potentially creating in flow starved regions were deposits could concentrate.

2. "Series 67"

The first group of plants that had a common design were called the "Series 67" or the ~2800 MWt Units. This category consists of Maine Yankee, Calvert Cliffs-1&2, St. Lucie-1, and Millstone-2. (See Figure XIV -a)

The Series 67 upper bundle design (see Figure XIV-b) consisted of 4 partial supports and a unibody design of the BW's and VS's. This unibody design meant that the VS's and BW's were welded together as one piece before being installed in the SG. The VS and BW material of the unibody was carbon steel. The width of each BW and VS was 4". The BW's laid across the tubes directly over the bend radius and came together to form a "V" design. The bottom of this "V" was tied directly to the topmost full eggcrate support. The VS's were not ventilated (i.e. did not have the elliptical flow holes punched through them).

The four partial support's consisted of 2 diamond patterned eggcrates below, and 2 drilled plates above. The top 2 partial supports were designed as drilled plates to provide a large amount of rigidity in the upper bundle (i.e. a very conservative design which had large margin of allowable stress).

The major change from the Early Units to the Series 67 Units was that spacing of the tubes in the VS region increased from 1.25" to 1.75" (which is the spacing still used in the newer generation of SG's). The spacing increase eliminated the use of the scalloped shape for the locking bars on the VS's (with the exception of Maine Yankee which has the scalloped bar design). The basis for the change was the observation in the field, that the scalloped shape locking bars were acting as crud traps. The flat locking bar had less crevice area than the half-moon shoe of the scallop bar. The significance of this change was the acknowledgment that crevices are undesirable (from either a corrosion or denting standpoint).

3. "3410 Series"

The next group of plants that had a common design were called the "3410 Series" or the 3410 MWt Units. Included in this category: ANO-2, Jersey Central (cancelled), SONGS-2&3, Waterford-2, and St. Lucie-2.

The 3410 Series upper bundle design (see Figure XIV -c.) consisted of the 3 partial eggcrates (PE's) and a segmented design of the BW's and VS's. This segmented design meant that the VS's and BW's were not joined together as one piece. The BW's were still one long piece, but the VS's were now each a single strip of carbon steel, leaving a gap between the BW and VS when viewed from the side. The width of each BW and VS was reduced to 2". The BW's were moved lower so that they were no longer laid across bend radius of the tubes. As a result, the bottom of the BW's no longer could come together to form a "V" design, since the point of the "V" would be too low in the bundle. Therefore, the BW's had a horizontal strip at the bottom called the "dogleg". The VS's were ventilated (i.e. had the elliptical flow holes punched through them for flow). The three PE's were all of the diamond pattern eggcrates design (i.e. no drilled plates). [Note that ANO-2 was a hybrid of the Series 67 and the 3410 Series designs. The unibody design was maintained, but the BW/VS unibody was raised higher to move the BW above the bend radius of the tubes (see Figure XIV -b.). The width of the BW's and VS's was reduced to 2" but the VS's were not ventilated. There were 4 PE's of the same design as the Series 67. Also, note the MWt rating: ANO-2 was <3410 MWt and of St. Lucie-2 was ~ 2800 MWt].

The basis for the design change was to minimize crevices/corrosion sites that had become evident in the Series 67 Units (and was an important issue for the industry in general at the time, as it is today). Thus:

a. It is noted that in the bend radius of a tube, the tube was oval from the 90 degree bend. This ovality puts the tube closer to the BW, causing a tight crevice. So, the BW's were moved lower, out of the bend radius, to increase the width of this crevice. This changed the "V" design and introduced the dogleg.

b. It was noted that the VS's in a unibody design must lay across a bend radius for some of the tubes. So, the VS's were disconnected from the BW's, creating the segmented design. As a result, no VS would lay across a bend radius of any tube.

c. The width of the VS's was reduced from 4" to 2", and eliminated 50% of the crevice length in the VS region.

d. The VS's were ventilated to allow some cross flow to "wash" the crevice sites and reduce the area of crevice sites in the VS region.

e. The PE's were reduced in number from 4 to 3. This eliminated 25% of the crevice sites in the PE region. The design of the drilled plate was eliminated which in turn eliminated the tight crevice sites in the PE region caused by the tolerances between the plate and the tubes.

Note: The changes to the tube support structure design were made possible due to the increased availability of tube support vibration data. Dynamic coefficients made it possible to predict accurate flow forces. Test data was also yielding realistic damping values for use in tube stability analysis. The availability of high speed computers with sophisticated structural and thermal hydraulic flow codes made it easier to not over-design the support structure and therefore reduce the number of areas with potential for flow starvation. However, the long lead time in SG materials and fabrication affected which Units could benefit from the new design changes.

The result of these design changes in the 3410 Series was that the upper bundle had more flexibility. Some of the design margin was used, when comparing the 3410 Series to the Series 67 design; however analytically the 3410 Series remained stable and conservative. Test data supported the analytical conclusions that the margins were acceptable against instability resulting from flow induced vibration.

While the tubes of the upper bundle were analytically stable, it became evident at SONGS that the new dogleg support portion of the BW was subject to flow induced static deflection. Investigation at SONGS lead to the discovery of high flow velocities in the central cavity. The central cavity of the CE U-tube SG's is basically empty due to the stay cylinder design. During power operation, that region is subject to higher flows. The flat strip of the BW dogleg was subjected to high cross flows, which resulted in the static deformation (i.e. static out-of-plane bending which in some cases resulted in plastic deformation). As a sail tacks against a high wind, the horizontal dogleg was statically forced into the adjacent tubes next to the central cavity, resulting in wear. This condition required that the innermost tubes around the central cavity be plugged on all of the 3410 Series Units (up to 150 tubes in some cases). This became known in the industry as the BW wear problem/phenomenon. It should be noted that to date, this is the only inherent design problem associated with the 3410 Series upper bundle support design.

4. "System 80"

The next group of plants that have a common design were called the "System 80" or the 3810 MWt Units. This category consists of Palo Verde 1,2,&3, Yellow Creek(TVA)-1&2, WPPSS-3&5, Duke Power-1 through 6, Boston Edison Pilgrim-2, and the Palisades replacement SG's (See Figure III -a). Of these Units, only Palo Verde (PVNGS) and Palisades is in operation; the others were cancelled (Note that the Palisades replacement SG's were just placed in operation 1992).

The principle reason for the design change from the 3410 Units was corrosion and manufacturing techniques (See Figure III -f):

a. The number of PE's was reduced from 3 to 2. This eliminated 33% of the corrosion sites in the PE region.

b. The BW and VS material was changed from carbon steel to 409-series stainless steel. All of the eggcrate material (full and PE's) was changed from carbon steel to 409-series stainless steel. This reduced the amount of surface oxidation so as to reduce the width of crevice sites throughout the entire bundle.

c. VS-2,4&6 were shortened and allowed to be "free floating". This was changed to assist manufacturing and also reduced the number of crevice sites in the VS region.

d. The number of I-beam overhead supports was reduce from three to two. This was an ease in fabrication and was considered to have no bearing on normal operating conditions.

The result of these design changes was that the System 80 upper bundle would have slightly more flexibility when compared to the 3410 Series. The design, however, was considered stable and conservative, from an analytical point of view.

Note 1:(The BW wear problem became known after the System 80 SG's were manufactured). It would therefore be assumed that the Batwing wear problem should have made itself evident during PVNGS power operation as the design of the System 80 BW is the same as the 3410 Series. However, other design changes (not related to the tube bundle supports) introduced an economizer section in the System 80 SG's. This design change introduced a flow distribution plate in place of the 01 support and a lower economizer feedwater nozzle. These design modifications changed the flow characteristics of the fluid in the central cavity such that the dogleg portion of the BW was no longer subject to flow induced wear.

Note 2: However, the economizer design of the System 80 was found to be subject to flow anomalies, not related to the design of the tube bundle support structure. The hotter downcomer recirculation flow was designed so as not to mix with the colder economizer flow entering the SG through the lower economizer nozzle. To keep the flow separated, a window was introduced in the tube shroud/wrapper plate above the economizer nozzle. The flow through this window was normal except near the divider plate; where the tube lane between the innermost row of tubes and the divider plate was of lower flow resistance than the rest of the bundle. The lower resistance meant high flow velocity caused flow induced vibration for those corner tubes that were closest to the divider plate and the wrapper plate. This resulted in the plugging of some of those tubes.

5. **"System 80 +"**

The latest group of Units designed by CE are those for the Korean Units that are currently under construction (i.e. Yangwong). The upper bundle support design from the System 80 was changed so as to incorporate the lessons learned from the System 80 and the 3410 Units:

a. The dogleg section of the BW was eliminated and the BW changed back to the original "V" design. This was to eliminate the concern of the BW wear problem that started at SONGS. As stated above, PVNGS does not exhibit this problem because of the economizer design. However, to be conservative, the Korean Units were changed to be sure that the BW wear problem would be eliminated.

b. When the BW was moved upward to form the "V", it was close enough to the VS's to be joined back to a unibody design. This meant that the concerns stated above were present again; namely the BW's and VS's would be located over the bend radius of the tubes. To compensate for the BW areas, the BW's were ventilated. For the remainder of VS's, it was deemed that this should not be such a concern as the VS's were also ventilated.

c. When the BW was moved upward to reform the unibody design, a third PE was added.

d. Note that the design of the downcomer recirculation window was also changed to eliminate the corner tube wear problem.

e. The hot leg flow distribution plate was eliminated as it was deemed not to impact blowdown or flow stability.

This design is more rigid than the PVNGS design. However, the original compromise between rigidity and corrosion sites should mean that this design would be slightly more subject to corrosion. Only when the on-line performance data is obtained can it be determined if this statement has any merit.

B. Description of Tube Examination Techniques

A detailed description and purpose of each tube examination method is described below.

1. <u>Receipt Inspections</u>

Radioactive tube sections are received by the laboratory and documented by the Health Physics technician. Tube sections are measured for length, checked for orientation and section markings.