ABB
ASEA BROWN BOVERI

February 12, 1992
LD-92-017

Docket No. 52-002

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

Subject: Response to NRC Requests for Additional Information

Reference: Letter, Plant Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 10, 1991

Dear Sirs:

The Reference requested additional information for the NRC staff review of the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC). Enclosure I to this letter provides our responses to a number of these questions including corresponding revisions to CESSAR-DC. Responses to the remaining questions of the Reference will be provided by separate correspondence.

Should you have any questions on the enclosed material, please contact 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
Acting Director
Nuclear Systems Licensing

vs/lw
Enclosures: As Stated

cc: J. Trotter (EPRI)
T. Wambach (NRC)

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RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION
PLANT SYSTEMS BRANCH

QUESTION 270.1

In response to RAI 270.1 parts a through h, a substantial amount of information that was provided in the March 15, 1991 response (LD-91-012) was not fully incorporated into the text of Section 3.11, for example the references to compliance with the requirements of 10CFR50.49 in response to RAI 270.1(f) has not been incorporated into the Amendment I version of the CESSAR-DC for the System 80+. The information provided in the responses to the RAIs should be fully incorporated into the text of the CESSAR-DC.

RESPONSE TO 270.1

C-E conducted a review of CESSAR-DC Section 3.11 and the March 15, 1991 responses (LD-91-012) to determine if any further information should be incorporated into CESSAR-DC. Listed below are the results of that review:

(1) Items (a) through (d-1) have been fully incorporated into Amendment I of CESSAR-DC;

(2) The following paragraph will be added to Section 3.11.3.2 to further incorporate C-E's response to item (d-2); note that lists identifying mechanical equipment components follows the procurement and qualification process: "Lists identifying the components of mechanical equipment and a bill of materials will be available in accordance with CENPD-255-A, Rev. 3";

(3) Item (d-3) has been fully incorporated into Amendment I of CESSAR-DC;

(4) Item (d-4) is covered by the response in (2) above;

(5) The following words will be added to Section 3.11.3.2 to further incorporate C-E's response to item (d-5): "The evaluation of environmental adequacy of equipment is initiated by the full definition of environmental requirements in equipment specifications, as stated above. Test reports and analyses which substantiate operability after exposure to the environment, and the quality assurance documentation, will be filed and available for staff audit, as discussed in Section 6.0, Documentation, of CENPD-255-A, Rev. 3 by the owner-operator";

(6) The following words will be added to Section 3.11.2.1 of CESSAR-DC to further incorporate C-E's response to item (e): "The detailed maintenance/surveillance program will be developed based on the specific equipment for that plant and the results of qualification testing and analysis for that equipment. This program is the responsibility of the owner-operator";

(7) The following words will be added to Section 3.11.2 to further incorporate C-E's response to item (f): "...and, as a result, System 80+ will comply with 10CFR50.49 consistent with CENPD-255-A, Rev. 3";

(8) Item (g) is not applicable to incorporation into

CESSAR-DC and item (h) has been fully incorporated into Amendment I of CESSAR-DC.

The as-stated changes above will be incorporated into a future Amendment of CESSAR-DC.

RAI 270.1

Chemical Spray

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

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

Typical values of chemical spray composition, concentration and pH are defined in Appendix 3.11A, Tables 3.11A-1, 3.11A-2 and 3.11A-3.

3.11.3 QUALIFICATION TEST RESULTS

3.11.3.1 Instrumentation and Electrical Equipment

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

3.11.3.2 Mechanical Equipment

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

Insert A → ¶

Insert B → ¶

3.11.4 CLASS 1E INSTRUMENTATION LOSS OF VENTILATION EFFECTS

The need for the HVAC systems and the design bases which prevent the loss of essential ventilation are described in Sections 6.4 and 9.4. In general, the two division concept of this plant provides 100% redundancy of all essential equipment and the HVAC system. In event of a failure of one system to deliver the

RAI 270.1

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

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

3.11.2 QUALIFICATION TESTS AND ANALYSES

Qualification of electrical equipment is in accordance with the program outlined in CENPD-255-A, Rev 3 (Reference 1). This document, written to comply with IEEE Std. 323-1974 and Category 1 of NUREG-0588, was accepted by the NRC Standardization & Special Projects Branch in September 1983. Insert D ~~The 1983 version introduces an alternate test profile which~~ allows for the substitution of testing twice at the specified service condition as a substitute for testing once at a profile which includes margins. For some equipment, System 80+ utilizes this option as an exception to IEEE Std. 323-1974. As written, CENPD-255-A applies to Combustion Engineering (C-E) supplied equipment. For System 80+ it is extended to include the BOP.

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

Equipment which, due to its location, is not significantly affected environmentally by the DBA is said to exist in a mild (normal plus abnormal service conditions) environment. The treatment for mild environment is taken from IEEE Std. 323-1983 rather than the 1974 version, which does not distinguish mild environment. For this equipment, if no significant aging mechanism at mild conditions can be identified a qualified life is not required. This applies to both electrical and mechanical equipment. If the predicted life based on experience, aging analysis, or tests is less than the design life of the plant, that equipment is subjected to a surveillance program and a preventative maintenance program that restores it to qualified operability. Insert C

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

RAI 270.1

The following INSERTS will be inserted into the appropriately marked CESSAR-DC Section:

INSERT A

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

INSERT B

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

INSERT C

The detailed maintenance/surveillance program for specific plants will be developed based on the specific equipment for that plant and the results of qualification testing and analysis for that equipment. This program is the responsibility of the owner-operator.

INSERT D

and, as a result, System 80+ will comply with 10CFR50.49 consistent with CENPD-255-A, Rev. 3.

QUESTION 270.2

a. The NRC staff has not accepted IEEE Std. 323-1983. Justify the use of the definition for a mild environment from this standard for the System 80+.

b. CENPD-255-A was originally accepted for use for the components identified as being supplied by Combustion Engineering. It is not readily apparent that the program outlined in this document can simply be extended to include the BOP. Provide a justification for the use of the program for BOP equipment and an explanation of how the program will be applied to the BOP.

c. In the discussion on Radiation For Harsh And Non-Harsh Environment Equipment reference is made to radiation above 10^4 Rads as the level for which equipment will be irradiated to its anticipated TID prior to type testing. This level should be 10^4 Rads and above.

d. Table 3.11B-1 uses descriptive terms to describe the required duration of operation during a design basis accident, such as continuous, short term, varies, and intermittent. Provide a more quantitative definition of these terms.

e. The meaning of the discussion of the testing procedure in Section 3.11.5.3 is not entirely clear. Clarify the procedure to be used to test equipment not subjected to a steam environment during DBA. Verify that the equipment is to be tested during the exposure to a high humidity environment rather than the equipment is to be subjected to a high humidity environment and subsequently tested.

f. In Section 3.11.5.4 reference is made to qualification by type test and/or analysis supported by partial type test data. 10 CFR 50.49 does not allow for qualification by analysis only. More fully describe what is meant by analysis and partial type test and show that this combination meets the requirements for testing and analysis allowed by 10 CFR 50.49.

RESPONSE TO 270.2

a. The definition of a mild environment is specified in IEEE 323-1983 as follows: "An environment expected as a result of normal service and extremes (abnormal) in service conditions where seismic is the only design basis event (DBE) of consequence." This definition is similar to the definition in CENPD-255-A, Rev. 3 for a Non-Harsh Environment which states that equipment located in a non-harsh environment will be qualified for the normal and abnormal local environment and a seismic event. There is no practical difference between the two definitions. The IEEE Standard does not introduce any new meanings to the definition for a non-harsh environment and, therefore, C-E believes that use of the IEEE 323-1983 definition for a mild environment is justified for System 80+.

b. CENPD-255-A, Rev. 3, Qualification of Class 1E Electrical Equipment, originally written to cover equipment within C-E's scope of supply, provides an acceptable methodology to meet the requirements of IEEE 323-1974 and the "Category I" requirements of NUREG-0588 to qualify Class 1E equipment for use in a nuclear power plant. This methodology can be applied to all Class 1E equipment. Therefore, whether the electrical equipment is supplied by C-E or others, the methodology to qualify all Class 1E electrical equipment for use in a nuclear power plant is the same.

c. The wording in CESSAR-DC Section 3.11.2.2 was taken from CENPD-255-A, Rev. 3, Section 3.4.2.1. The value of 10^4 Rads has been determined to represent the absorbed dose level above which the effects of radiation on materials must be analyzed. 10^4 Rads is, however, considered to be a level well below the radiation damage threshold of materials. The radiation damage threshold is that level at which a change in material property can first be detected. For conservatism, if a threshold value is defined as less than 10^4 Rads, the value of 10^4 Rads is used as the upper bound to envelope all conditions. Therefore, damage would be expected and equipment would be tested only when the absorbed dose is above 10^4 Rads.

d. A more quantitative definition of the descriptive terms used in Table 3.11B-1 of CESSAR-DC to describe the required duration of operation during a design basis accident is as follows:

continuous - the specific component is required to operate throughout the design basis accident without interruption (i.e. up to six months);

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

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

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

e. When equipment is not subjected to a steam environment during a DBA but is required to operate in a high humidity environment, its operability in that high humidity is demonstrated by test. The equipment is tested prior to the application of the high humidity environment (baseline), tested while exposed to the high humidity environment to prove operability and again tested after removal of the high humidity environment (baseline). The comparison of the baseline tests determine if any degradation is present and to meet failure criteria that equipment must operate prior to, during and after an event (condition).

f. C-E makes no claim in Section 3.11.5.4 in CESSAR-DC that equipment qualification for submergence is by analysis only. However, C-E does state in Section 3.11.5.4 that qualification for submergence may be demonstrated by analysis supported by partial type test data. This meets the 10 CFR 50.49 criteria.

The qualification analyses are described in Section 5.2.1 of CENPD-255-A, Rev. 3.

NRC Question 281.34

The response to RAI 281.34 indicated that Section 9.1.3.3 would be revised to include part of the response. Amendment I of Section 9.1.3.3 does not include the revision.

Response to NRC Question 281.34

It is intended that the next amendment to CESSAR-DC will include the sampling and pool purification demineralizer resin replacement criteria stated in our previous response. This information will be included in Section 9.1.3.2.2.5 and 9.1.3.3.3 of CESSAR-DC. Revised Sections are included for information.

Question 410.54

The response to RAI 410.54 is incomplete. Please provide the values for the maximum lifting height assumed for each case analyzed.

Response 410.54

Lift heights and grapple weights are not known prior to final procurement of the refueling machine used in the fuel building. System 80+ spent fuel racks can absorb an impact energy of 93,100 inch-lbs without exceeding the rack K_{eff} criteria. The refueling equipment will be designed so that the maximum impact energy resulting from a dropped fuel assembly and handling tool, a dropped handling tool or any other dropped fuel handling related load from their maximum respective lift heights will not exceed the energy absorbing capacity of the fuel rack while maintaining K_{eff} criteria. In addition, administrative controls will limit the size and lift height of any other non fuel handling loads that are carried over the fuel racks such that this maximum impact energy is not exceeded.

Question 410.55(a):

The response to RAI 410.55(a) is considered not adequate. Provide, as a minimum, the heat removal rates required to meet the design bases criteria for normal and abnormal conditions, and the design heat removal rates for these conditions.

Response 410.55(a):

The heat removal rates required to meet the design bases criteria for normal and abnormal conditions and the design heat removal rates for these conditions are discussed in the response to NRC Question 410.56.

Question 410.55(d):

The response to RAI 410.55(d) is considered not adequate. It did not respond to items (4) and (5).

Previous Question 410.55(d):

SFPCCS design provisions to maintain acceptable pool water conditions per SRP 9.1.3, Section III.7 guidance in the following areas:

(4) refueling canal coolant processing ability, and

(5) features to prevent the inadvertent transfer of spent filter and demineralized media to any place other than the radwaste facility.

Response 410.55(d) (4):

CESSAR-DC Section 9.1.3.3.3 "Water Quality" states that the design flow rate and filtering capability of the PCPS shall be such that the refueling pool water chemistry and clarity are sufficient for an operator to read fuel assembly identification numbers that are 3/8 inches high, 3/16 inches wide and 1/16 inches thick from the refueling machine at the time the operators and refueling equipment are ready to move fuel (i.e., designed such that water clarity problems do not cause refueling delays).

Since the refueling canal is directly integrated with the refueling pool, the PCPS piping located in the pool will be orientated to ensure the proper circulation of pool water and refueling canal water to meet the above requirement.

Response 410.55(d) (5):

CESSAR-DC PCPS Figure 9.1-3 shows a filter in each purification loop. System 80+ PCPS filters require replacement and are not backflushable. Therefore, there is no possibility of inadvertant transfer of spent filter media or trapped particles into the process fluid.

The PCPS filter units are designed for remote replacement of the filter cartridge. Since the filters are bottom loaded when filter replacement is required the filter is isolated from the process fluid and the lines are drained to the CVCS recycle drain header. The entire filter cartridge is removed from the area for SWMS processing. A new filter is loaded into the housing, and process flow is re-established.

CESSAR-DC PCPS Figure 9.1-3 shows a pool ion exchanger in each purification loop. The ion exchangers in the Pool Cooling and Purification System are the same design as those employed in the Chemical and Volume Control System. Each ion exchanger contains a flow distributor on the influent to prevent channeling of the resin bed and a resin retention element on the effluent to preclude discharge of resin with the effluent process fluid. These units are provided with connections required to replace resins by sluicing to the SWMS.

Response 410.55(d) (5) (Cont'd):

Detailed operating requirements, along with system design provisions, are provided for PCPS ion exchanger replacement to ensure that no resin fines are transferred to any place other than the SWMS. During resin replacement, the ion exchanger is bypassed and then isolated from the process fluid. Next the ion exchanger is depressurized via the vent line and drained to the ion exchanger drain header of the Chemical and Volume Control System. There is a "Y" type strainer in this line located to retain any resin that might have escaped during the draining process. If strainer loading is indicated (via pressure instrumentation), the dedicated line running from the "Y" section of the stainer directly to the SWMS is opened to direct the solid waste to the SWMS. This configuration ensures that only resin free fluid is introduced into the equipment drain tank. A realignment is then made for resin sluicing to the SWMS. Detailed operating procedures ensure that the resin will not be inadvertantly discharged to anywhere but the SWMS. When sluicing is completed, the ion exchanger is reconfigured to the process fluid flowpath.

Strainers are also located in the PCPS downstream of each ion exchanger. These strainers are also the "Y" type and are located to prevent any resin from entering the spent fuel pool or refueling pool due to ion exchanger lower retention element failure. Again, dedicated lines run from the "Y" section of the strainers directly to the SWMS assuring resin will be directed to the radwaste facility. Strainer loading is indicated via differential pressure instrumentation.

Question 410.56:

The response to RAI 410.56, regarding heat generation rate calculations is considered incomplete. Provide the residual decay heat release-vs-time curves generated with the ORIGEN 2 methodology to substantiate the statement of conservatism. Provide the resulting heat loads that need to be removed and a comparison of these heat loads to the design heat removal rates for the system.

Response 410.56:

The attached Figure shows the representative fuel decay energy versus time after shutdown calculated with the ORIGEN 2 methodology for a high average discharge burnup (44,000 Mwd/t). The results shown indicate uncertainties based on ANSI/ANS - 5.1, "American National Standard for Decay Heat Power in Light Water Reactors," 1979. In addition, an uncertainty of 10% is applied to the results in thermal-hydraulic analyses of the spent fuel pool, including calculation of the normal and maximum heat loads.

The normal heat load on the spent fuel pool is calculated assuming that the pool contains an accumulation of spent fuel equal to 10 full power years, with the newest spent fuel batch having just been placed in the pool during refueling at 120 hours after shutdown. The resultant heat load for that condition is 2.7×10^{7} BTU/hr. During a full core offload, it is assumed that 1 full core is placed in the spent fuel pool 120 hours after shutdown. In addition, the pool is assumed to already contain

RAI 410.56 (Cont'd)

the normal heat load as described above. The resultant heat load for this case is 7.36×10^7 BTU/hr, which is the maximum heat load for the spent fuel pool. The heat load from any other combinations of spent fuel within the pool will result in heat loads lower than 7.36×10^7 BTU/hr. The design heat removal rate of the Pool Cooling and Purification System is 7.51×10^7 BTU/hr.

RAI 410.56

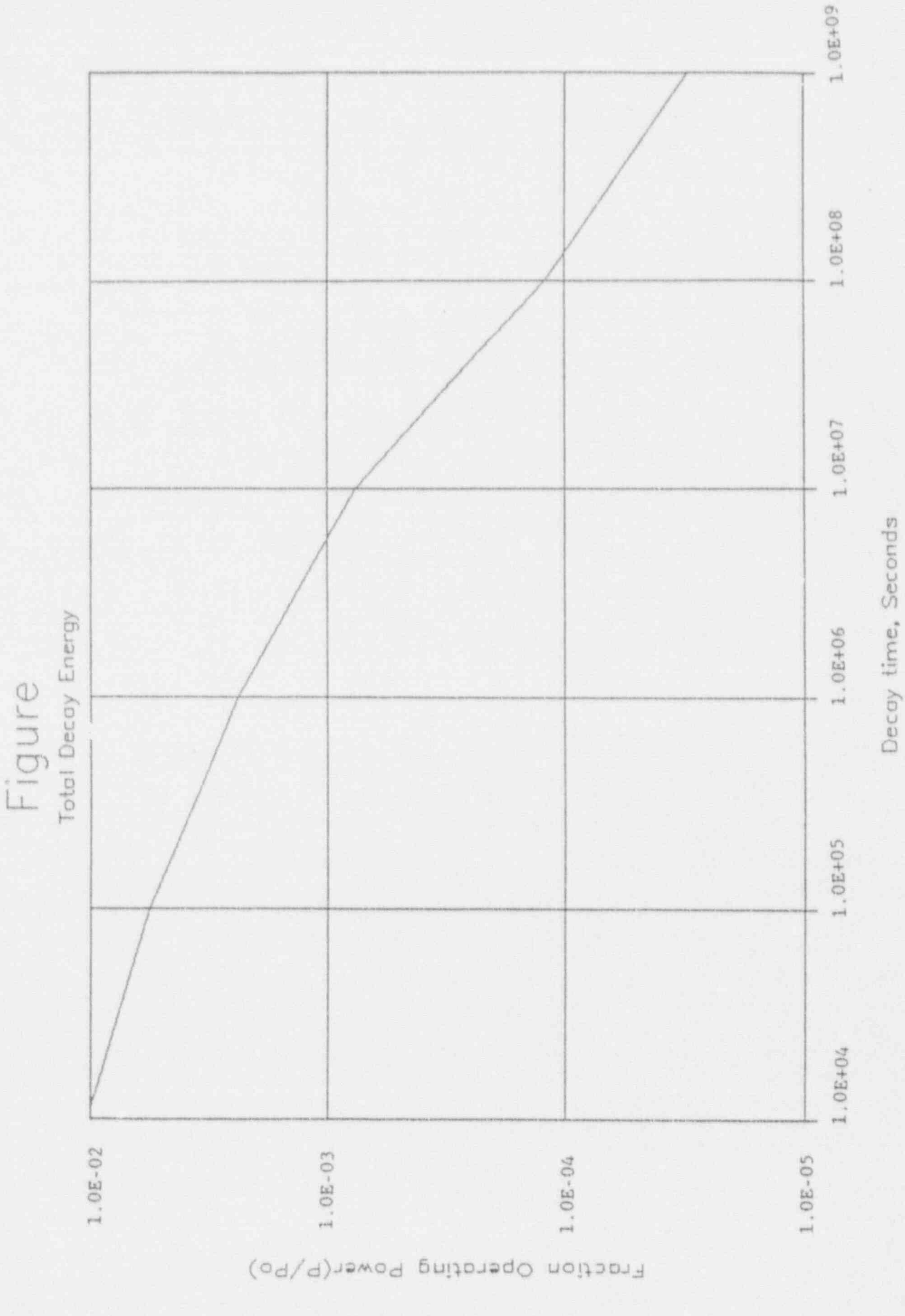

Figure
Total Decay Energy
1.0E-02
1.0E-03
1.0E-04
1.0E-05
Fraction Operating Power(P/Po)
1.0E+04
1.0E+05
1.0E+06
1.0E+07
1.0E+08
1.0E+09
Decay time, Seconds

Question 410.59

The response to RAI 410.59 is considered not adequate. The description of the system in Section 9.1.3 does not contain the details of the information requested, and the design bases, as stated in Section 9.1.3, are not specific enough to ensure that the design will satisfy the requirements stated. Include, in terms of design bases or design description, the commitment that the following general features will be included in the design:

- A leakage detection system to detect component or system leakage.
- Components and headers of the system to be designed to provide individual isolation capabilities to assure system function, control system leakage, and allow system maintenance.
- Design provisions to be included to assure the capability to detect leakage of radioactivity or chemical contamination from one system to another and to preclude long term corrosion, organic fouling or the spread of radioactivity.

Response to Question 410.59

Section 9.1.3 of CESSAR-DC will be amended to address the design information noted above. Section 9.1.3 of CESSAR-DC will include the following:

The Pool Cooling and Purification System (PCPS) design includes the following features:

- A means for detecting leakage from the system (or components of the system).
- Components and headers designed to provide individual isolation capability to assure system function, control system leakage, and allow system maintenance.
- A means for detecting radioactive leakage and chemical contamination from interfacing systems, and the ability to preclude the long term effects of chemical contamination or the spreading of radioactivity.

CESSAR-DC and the System 80+ PCPS P&ID (Figure 9.1-3) presently include features required in Standard Review Plan 9.1.3, Section III.3.

- Component or system leakage from the PCPS is detected by several means, including area sump and floor drain level monitoring, Chemical and Volume Control System (CVCS) equipment drain tank level monitoring, Radiation Monitoring System (RMS) radiation monitors, spent fuel pool level monitoring, and, during refueling operations, refueling pool level monitoring. The auxiliary building floor drains are provided with adequate capacity for anticipated leakage and their levels are monitored. The CVCS equipment drain tank is designed to receive leakage from PCPS valves and components. The tank is used to monitor PCPS leakage. High level is alarmed in the control room. RMS radiation monitors in the containment and auxiliary buildings are provided to detect radioactive system leakage and are alarmed for high or increasing radiation levels. A high leakage rate is detected by a change in spent fuel pool level, and, during refueling operations, a change in refueling pool level. Both pools have low level alarms in the control room.

- The PCPS is composed of two independent cooling trains used to remove heat from the spent fuel pool and two independent purification trains used to purify water in the spent fuel pool, and, in refueling operations, the fuel canal and refueling pool. In order that system function may be assured, system leakage controlled, and system maintenance performed, all components and headers in this system are capable of being individually isolated. To prevent a loss of system function when individual components are isolated, the independent trains are cross-connected. Normally closed valves are opened to allow the cooling or purification flow to bypass the isolated components in a train and flow through the identical components in the redundant train. If required, an entire train can be isolated and the redundant train used to fulfill system requirements. In addition, the filters and ion exchangers in the system can be isolated and bypassed without the need to divert flow to the second train. System leakage can be controlled by isolating the leak and rerouting flow through the redundant train so that the required cooling or purification function remains available. System maintenance is performed by isolating the equipment needing service and bypassing flow to the redundant train. If necessary, an entire train can be isolated and shut down for limited periods of time while the second train performs the required system functions.

Intersystem leakage is detected in several ways. PCPS leakage to the Component Cooling Water System (CCWS) through a failure of the spent fuel pool cooling heat exchangers is detected by radiation monitors present in the CCWS which detect PCPS to CCWS leakage. Leakage to the CVCS is detected by monitoring the equipment drain tank level. Leakage from the PCPS to the Solid Waste Management System is detected by changes in spent resin storage tank levels. Leakage to the Incontainment Refueling Water Storage Tank (IRWST) from the PCPS is detected through monitoring of the spent fuel pool, refueling pool, and IRWST levels. Leakage to the PCPS from other systems is detected by changes in spent fuel pool and refueling pool levels and analysis of samples taken from the PCPS. PCPS filters and ion exchangers preclude long term corrosion, organic fouling, and the spread of radioactivity in the system.

by use of the system's skimmers. The cleanup system is designed for a flow rate sufficient to ensure adequate circulation of the entire spent fuel pool water volume, and to maintain the specified water chemistry.

The boron concentration in the spent fuel pool water is maintained at approximately the same concentration as in the refueling water. Provisions are made to make up water to the spent fuel pool. The makeup water meets all specified water chemistry requirements.

9.1.3.1.4 System Capacity Bases

For all normal plant operations and normal spent fuel pool heat load conditions, the maximum spent fuel pool bulk water temperature is 120°F. Under heat load conditions of spent fuel in all usable rack spaces (Section 9.1.2.2.2), which includes, as a minimum, a full core offload with 10 years of irradiated fuel in the pool, the maximum bulk water temperature is 140°F. Given a single active failure, the maximum temperatures for normal conditions or a full core offload are 140°F or 180°F respectively. The normal heat load is the decay heat which occurs when an accumulation of spent fuel equal to 10 full power years is in the spent fuel pool, with the newest spent fuel batch having just been placed in the pool during refueling at 120 hours after shutdown. The full core offload heat load is equal to the normal heat load plus the addition of the decay heat from a full core offload 120 hours after shutdown. Design heat loads are evaluated utilizing ANSI/ANS 5.1 (proposed version approved by Subcommittee ANS-5 ANS Standard Committee, October 1971) decay heat correlations. I

A Seismic Category I, Quality Group B borated makeup water source is provided to the spent fuel pool. Nonborated water from a non-seismic source is used to make up for the evaporation losses from the spent fuel pool during normal operation.

INSERT A

9.1.3.2 System Description

9.1.3.2.1 General Description

The safety-related spent fuel pool cooling system consists of two independent cooling trains. The system is located in a Seismic Category I building which provides protection from the effects of natural phenomena and missiles. The spent fuel pool cooling system (piping, pumps, valves, and heat exchangers) is safety-related, Quality Group C. The spent fuel pool receives

RAI 410.59

INSERT A

9.1.3.1.5 Leakage Detection and Isolation Capabilities

The Pool Cooling and Purification System (PCPS) design includes the following features:

- A means for detecting leakage from the system (or components of the system).

- Components and headers designed to provide individual isolation capability to assure system function, control system leakage, and allow system maintenance.

- A means for detecting radioactive leakage and chemical contamination from interfacing systems, and the ability to preclude the long term effects of chemical contamination or the spreading of radioactivity.

Question 410.61:

Provide an evaluation that assures that any failure in the nonsafety-related spent fuel pool cleanup and associated systems cannot affect the functional performance of any safety-related components in accordance with SRP 9.1.3, Section III.5 guidance. Your response by submittal dated May 15, 1991 did not adequately address this question. The referenced P&ID does not provide sufficient information to determine the affects that a failed nonsafety-related system or component will have on a safety-related component.

Response 410.61:

ABB-CE has evaluated the non-safety related spent fuel pool cleanup and associated systems with respect to failure effects on safety-related systems and finds the following:

The pool purification system of the pool cooling and purification system P&I Diagram, Figure 9.1-3, shows that its flow paths are independent of other systems. Its failure to operate will not affect the functional performance of the safety-related spent fuel pool cooling system (SFPCS).

Pool purification system components (including electrical components) are physically separated from SFPCS components to an extent which ensures that their failure will not reduce the performance of the SFPCS. Refer to the general arrangement drawings.

The pool purification system incorporates several interfaces as shown on the reference P&I Diagram. The following interfaces are all nonsafety-related and are also isolated with normally shut gate valves:

- Chemical and Volume Control System
- Solid Waste Management System
- Building Vents
- Various Local Sampling Stations

The remaining interfaces are as follows:

- The pool purification system inlet and outlet lines of the refueling pool pass through the containment building. Here the pool purification system incorporates isolation valves on either side of the containment. The isolation valves and associated piping are safety class 2 and are designed to ASME B&PV code Section III, Subsection NC rules.

- The pool purification system lines penetrate the safety-related spent fuel pool in several locations. Each line penetration incorporates a normally open gate valve. This allows for isolation from the fuel pool if necessary.

Response 410.61 (Cont'd):

- Downstream of the refueling pool ion exchanger 1, the pool purification system has a connection that interfaces with the IRWST via the safety injection system. Here the pipe goes through two class changes and two normally shut isolation gate valves.

- The refueling pool drain lines run directly to IRWST. These areas are isolated from each other through the use of normally shut gate valves. If a failure occurs that partially prevents refueling water from being drained back to the IRWST, it will not be a safety concern since the reactor is shut down at this time.

- Pool purification system piping in the spent fuel pool is arranged so that failure of any one pipe line cannot drain the spent fuel pool below the water level required for shielding. This arrangement assures that a failure in the pool purification system does not degrade the function of the SFPCS to ensure coolant medium availability.

The above information concludes that a nonsafety-related pool purification system component failure will not affect the functional performance of any safety-related components.

Question 410.64

a. The application does not include the required discussion regardir ; failure of non-safety-related systems and structures and their potential effects on the integrity and coolability of the spent fuel racks. Provide this analysis.

b. The response to RAI 410.64 is adequate with the following exception: The response commits to add an insert to Section 9.1.2.2.2 in the next revision of the Section. Amendment I of the Section does not contain the insert.

Response 410.64

a. All building components and structures in the area of the spent fuel pool are designated Seismic Category II to preclude their failure and entry into the spent fuel pool during a seismic event.

b. This omission will be corrected in a future submittal.

RAI 410.64

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The following design bases are imposed on the storage of fuel within the spent fuel pool:

A. Accidental criticality shall be prevented for the most reactive arrangement of fuel stored with optimum moderation by avoiding a K_{eff} greater than 0.95. This design basis shall be met under any normal or accident conditions. E

B. The requirements of Regulatory Guide 1.13 shall be met.

C. The storage racks and facilities shall be Seismic Category I.

D. Storage shall be provided for up to 907 spent fuel assemblies. I

9.1.2.2 Facility Description

9.1.2.2.1 Spent Fuel Pool

The spent fuel pool is a stainless steel lined, concrete walled pool that is an integral part of the fuel building. E

9.1.2.2.2 Spent Fuel Pool Storage Racks

The spent fuel pool storage racks are made up of twelve 11x11 individual modules containing 121 storage cells each (see Figures 9.1-21 and 9.1-22). I A module is an array of fuel storage cells similar to that shown in Figure 9.1-1. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells. The stainless steel construction of the racks is compatible with fuel assembly materials and the spent fuel borated water environment. E

Delete ~~A single pitch of 9.780 inches is provided for all of the racks in the pool.~~ The spent fuel is stored in two regions of the pool. Region I provides core off-load capability for 363 spent fuel assemblies (equivalent to one and one-third cores). I This is achieved with 50% density storage in a checkerboard array using "L" inserts in the usable cells (Figure 9.1-2). The "L" insert is a non-poisoned stainless steel insert which provides the needed flux trap water gap. E Region II provides 75% density storage for 544 spent fuel assemblies. The cells that are not used are blocked to prevent improper storage. A total of 907 usable spaces for spent fuel storage is thus provided. I

The structural design of the spent fuel rack and pool includes provisions for accepting loads associated with 100% storage with neutron poison inserts in order to meet future expansion potential.

Insert

"A fuel assembly may be stored in Region II only if it has the minimum burnup required for an assembly of its initial enrichment. The Owner-Operator will develop and implement administrative controls to permit storing a fuel assembly in Region II only if it meets established burnup versus initial enrichment requirements."

Question 410.67:

The spent fuel pool cooling system must be designed with suitable redundancy of components that safety functions can be performed assuming a single failure of a component coincident with the loss of all offsite power. Your submittal does not provide the information necessary to verify that the system can continue to perform its intended function without offsite power. Provide the failure modes and effects analysis (FMEA) to verify that the system is capable of meeting this requirement.

Response 410.67:

Section 9.1.3.3.1, Availability and Reliability, states that in the event of the failure of a pool cooling pump or loss of cooling to a heat exchanger, the second cooling train provides backup capability thus assuring continued cooling of the spent fuel pool.

Section 8.3, Onsite Power System, contains Tables 8.3.1-2 and 8.3.1-3 which list the spent fuel pool cooling pumps as being capable of operation through the use of Class 1E onsite power. Also listed in these tables are the component cooling water pumps which assure that cooling water will be supplied to the spent fuel pool cooling system heat exchangers during this period.

Based on the above information, a pool cooling train will be available to assure pool cooling assuming a single failure of a component coincident with the loss of offsite power.

A failure modes and effects analysis (FMEA) on the system will be included in a future Amendment to CESSAR-DC as Table 9.1-3.

The PCPS has its maximum duty during refueling operations, when the decay heat from the spent fuel is the highest (see Section 9.1.3.4) and water clarity is required to facilitate refueling operations. The system is normally placed in operation prior to the transfer of any fuel, and is continued in operation as long as required to maintain temperature and/or water purity.

9.1.3.3 Safety Evaluation

9.1.3.3.1 Availability and Reliability

The PCPS has no emergency function during an accident. A cooling train may be shut down for limited periods of time for maintenance or replacement of malfunctioning components. In the event of the failure of a pool cooling pump or loss of cooling to a heat exchanger, the second cooling train provides backup capability, thus assuring continued cooling of the spent fuel pool.

9.1.3.3.2 Spent Fuel Pool Dewatering

The most serious failure of this system would be complete loss of water in the spent fuel pool. To protect against this possibility, the spent fuel pool cooling pump suction connections enter near the normal water level, so that the pool cannot be gravity drained. The return lines contain an antisiphon hole to prevent the possibility of gravity draining of the pool via these lines. I

The CVCS provides a manual makeup capability of borated water to the PCPS refueling pool and spent fuel pool sufficient to make up for a 100 gpm leakage rate out of the spent fuel pool.

9.1.3.3.3 Water Quality

Only a very small amount of water is interchanged between the refueling canal and the spent fuel pool as fuel assemblies are transferred in the refueling process. Whenever a fuel assembly with defective cladding is transferred from the fuel transfer canal to the spent fuel pool, a small quantity of fission products may enter the spent fuel pool water. The cleanup loops remove fission products and other contaminants from the water. Radioactivity concentrations will be maintained such that the dose at the surface of the spent fuel pool will be 2.5 Mrem/hr or less.

A Failure Modes and Effects Analysis for the Spent Fuel Pool Cooling System is presented in Table 9.1-3.

Table 9.1-3
(Sheet 1 of 2)

Failure Mode Analysis of the Spent Fuel Pool Cooling System

Component	Failure Mode/Cause	Effect on System	Method of Detection	Inherent Compensating Provision
SFC Pumps	Pump inoperable/ mechanical or electrical failure	Loss of flow in one division	Motor status in control room, local flow indication, high temperature alarm in control room or local temperature indication.	Redundant division is provided for continued flow for heat removal
SFC Heat Exchanger	Heat exchanger malfunction/Blockage	Loss of heat removal in one division	High temperature alarm in the control room or local temperature indication	Redundant division is provided for continued heat removal
	Heat exchanger malfunction/Leaking tubes	Reduced flow in one division	Local flow indication	Redundant division is provided for continued heat removal
Valves	Pump suction valve fails closed/operator error or mechanical binding	Loss of flow in one division	Local flow indication or low pressure alarm in the control room	Redundant division is provided for continued heat removal
	Pump discharge valve fails closed/operator error or mechanical binding	Loss of flow in one division	Local flow indication or low pressure alarm in the control room	Redundant division is provided for continued heat removal
	Heat exchanger isolation valve fails closed/operator error or mechanical binding	Loss of flow in one division	Local flow indication	Redundant division is provided for continued heat removal

Table 9.1-3 (Cont'd)
(Sheet 2 of 2)

Failure Mode Analysis of the Spent Fuel Pool Cooling System

Component	Failure Mode/Cause	Effect on System	Method of Detection	Inherent Compensating Provision
Piping (pipe break)	Loss of pump suction line/linebreak or mechanical damage	Loss of flow in one division	Local flow indication or low pressure alarm in the control room	Redundant division is provided for continued heat removal
	Loss of pump discharge line/linebreak or mechanical damage	Loss of flow in one division	Local flow indication or low pressure alarm in the control room	Redundant division is provided for continued heat removal
	Loss of return line/linebreak or mechanical damage	Loss of flow in one division	Local flow indication or low pressure alarm in the control room	Redundant division is provided for continued heat removal

Question 410.68

The response to RAI 410.68 includes the statement that "the statement made in Section 9.1.3.3.1 has been corrected in Amendment I." Amendment I still contains the statement that the PCPS has no emergency function during an accident.

Response to 410.68

Amendment I to CESSAR-DC Section 9.1.3.3.1 contains a statement which reads:

"The PCPS has no emergency function during an accident."

The purpose of this statement is to document that the PCPS is not credited for recovery from any Chapter 15 design basis accident. Reiterating our previous response (of March 26, 1991) the PCPS is designed in accordance with GDC 44 to transfer heat from the spent fuel pool, under normal and accident conditions.

Accordingly, Section 9.1.3.3.1 will be reworded in the next Amendment to CESSAR-DC as follows:

9.1.3.3.1 Availability and Reliability

The safety function of the PCPS is to transfer heat from the spent fuel pool to the Component Cooling Water System according to the design parameters established in Section 9.1.3.1. This function is achieved under both normal and accident conditions. Suitable redundancy is provided to ensure that this function can be achieved assuming a single failure of a component coincident with the loss of either onsite or offsite power. In the event of a failure of a pool cooling pump or loss of cooling to a heat exchanger, the second cooling train provides backup capability, thus assuring continued cooling of the spent fuel pool. A cooling train may be shut down for limited periods of time for maintenance or replacement of malfunctioning components.

Question 410.101:

As stated in Section 5.2.5.1.1.3 of the System 80+ CESSAR, "the particulate monitoring system is capable of functioning when subjected to an SSE." However, this system is not specifically identified in Table 3.2-1, Classification of Structures, Systems and Components. Therfore, clarify which system in Table 3.2-1 is the "particulate monitoring system" desig monitor RCPB leakage.

Response 410.101:

The Containment Atmosphere Monitor is used to detect leakage of reactor coolant by measuring particulate and gaseous beta activity. The Containment Atmosphere Monitor is discussed in Section 11.5. Section 5.2.5.1.1.3 is being revised to correctly identify this monitor. Table 3.2-1, currently being revised in response to RAI 210.1, accurately reflects the name of this monitor and its seismic classification. Table 11.5-3 is also being revised to be consistent with the response to RAI 210.1.

RAI 410.101

5.2.5.1 Leakage Detection Methods

5.2.5.1.1 Unidentified Leakage

The methods employed to detect unidentified leakage are presented in the following sections. I

5.2.5.1.1.1 Inventory Method

Leakage from the Reactor Coolant System (RCS) can be determined by net level changes in the pressurizer and in the volume control tank since the RCS and the Chemical and Volume Control System (CVCS) are closed systems. Since letdown flow and the reactor coolant pump seal controlled bleedoff flow are collected and recycled back into the RCS by the CVCS, the net inventory in the RCS and CVCS under normal operating conditions will be constant. Transient changes in letdown flow rate or RCS inventory can be accommodated by changes in the volume control tank level. Makeup flow rate provides a means of detecting leakage from the RCS through measurement of the net amount of makeup flow to the system. The net makeup to the system under no-leakage steady state conditions should be zero. The makeup flow rates and the integrated makeup flow from the Reactor Makeup System are continuously monitored and recorded. Analysis of the integrated makeup flow recorders over a period of steady state operation can provide detection of abnormal leakage. An increasing trend in the amount of makeup required will indicate an abnormal leak which is increasing in rate. Leaks occurring suddenly will be indicated by a step increase in the amount of makeup which does not decrease as would be the case for a purely transient condition. I

5.2.5.1.1.2 Sump Level & Flow Method

Leakages are routed to the holdup volume sumps, or reactor cavity sump. Sump levels are monitored in the control room and alarmed on high level. Leakage rate is calculated utilizing the change of sump level. Additionally, the amount of leakage is calculated utilizing the frequency and lengths of time the sump pumps operate in combination with the known flow rate of the pumps. This system addresses Regulatory Guide 1.45 Position 2, 3(1). I

5.2.5.1.1.3 Containment Air Particulate and Gaseous Radioactivity Monitoring

The Containment Atmosphere monitor takes continuous containment air samples and measures the particulate and gaseous beta activity. High alarms are displayed in the control room. Leakage of reactor coolant has the effect of increasing radioactive particles and gas in the containment, thus an increase in activity above base line is a direct indication of leakage. Conversion to leakage rate is discussed in Section 5.2.5.2.4.1. Radiological monitoring is discussed in more detail in Section 11.5.

The ~~particulate monitoring system~~ Containment Atmosphere Monitor is capable of functioning when subjected to an SSE.

5.2.5.1.1.4 **Acoustic Leak Monitoring System (ALMS)** I

The ALMS is fully described in Section 7.7.1.6.2. In summary, it is a system which monitors changes in acoustic levels above a normal background with no leakage. There are 9 locations where sensors are located (see Table 7.7-3). The locations were selected based on the criticality of the component or region, and in some cases (such as manways) where leakage has been experienced in the industry. The indication is qualitative; it alerts operators to investigate and compare with other leakage monitoring methods.

5.2.5.1.2 **Identified Leakage**

The amount of identified leakage from the Reactor Coolant System can be determined by adding up the amounts from all identified paths described below. Indicators and alarms associated with all of the identified leakage paths are provided in the control room.

5.2.5.1.2.1 **Valves Located on the Reactor Coolant System**

The primary safety valves and safety depressurization valves, located at the top of the pressurizer are routed to the In-Containment Refueling Water Storage Tank (IRWST). Each valve is monitored for seat leakage by an in-line Resistance-Temperature Detector (RTD).

Positive indication of valve leakage will be provided in the control room. Monitoring will be provided by an Acoustic Leak Monitoring System (ALMS) consisting of an accelerometer (acoustic sensor) mounted downstream of each valve. A plant annunciator alarm will be provided to alarm if the valve is not fully closed. The ALMS is part of the NSSS Integrity Monitoring System (see Section 7.7.1.6.2 for a detailed description). I

5.2.5.1.2.2 **Reactor Coolant Pump Seals**

Instrumentation is provided to detect abnormal seal leakage. The reactor coolant pumps are equipped with two-stage seals plus a vapor or backup seal as described in Section 5.4.1.2. During normal operation, the reactor coolant system pressure is

TABLE 11.5-3

(Sheet 1 of 2)

AIRBORNE RADIATION MONITORS

Monitor (-channel)	Detector Type[a]	Typical Sensitivity (μCi/cc)	Typical Range (μCi/cc)	Power Supply	Seismic Category[b]	Automatic Function[c]
Containment Atmosphere						
- Particulate	Beta	7E-12 (Cs-137)	1E-11 - 1E-5	1E	~~None~~ I	None
- Iodine	Gamma/SCA	3E-11 (I-131)	1E-12 - 1E-6	Non-1E	None	None
- Gas	Beta	5E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None
Radwaste Building Ventilation						
- Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None
Fuel Building Ventilation						
- Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	Divert Flow
Ventilation Systems Multisampler (12 pts.)						
- Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None
Control Room Air Intake (2 monitors)						
- Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	1E	I	Isolate Intake

(a) "Beta" = Beta Scintillation Detector
"Gamma" = Gamma Scintillation Detector
"SCA" = Single Chemical Analyzer

(b) Final seismic category determination for all monitoring equipment will depend on final equipment layout and non-seismic interaction considerations.

(c) Automatic Functions for Airborne Monitors are described in Section 11.5.1.2.4.

I

NRC RAI 410.101

Question 410.102a

In Section 6.8.3 of the CESSAR the statement is made that the instrumentation requirements for the in-containment refueling water storage tank (IRWST) are described in Section 7.4.1.3. However, this section was not provided in Amendment I. Therefore, provide information on the instrumentation requirements for the operation of the IRWST, including level, temperature, and pressure indication and alarms.

Response 410.102a

The IRWST water level is monitored with two 1-E channels and is displayed in the Control Room with a range of 0 to 100 percent. High and low level alarms are provided.

The IRWST water temperature is monitored with two 1-E channels and is displayed in the Control Room with a range of 50 to 250 degrees Fahrenheit. There are no temperature alarms.

The IRWST pressure is not monitored since it is essentially the same as the containment.

6.8.2.2.2 **Steam Relief System**

The function of the Steam Relief System (SRS) is to transport steam or water relieved through the Pressurizer Safety Valves (PSVs) and Rapid Depressurization Valves (RDVs) to the IRWST and dissipate its thermal energy. INSERT A

The SRS is comprised of the following components (Figure 6.8-3):

A. Two discharge lines from the PSVs and RDVs to headers in the IRWST.

B. Two headers in the IRWST connecting the discharge lines to the sparger supply lines.

C. Six sparger supply lines per header connecting the headers to the spargers.

D. Load reduction devices in each sparger supply line to reduce the loads on the SRS and the IRWST during initial stages of the steam discharge process.

E. Twelve sparger heads to maintain stable condensation of the steam and provide for effective mixing of the condensed steam with the IRWST water. Sparger heads are cylindrical sections containing a hemispherical head at the end of the sparger supply lines. I

Upon actuation of one or more PSVs or RDVs, steam or water flows through the main discharge lines to distribution headers in the IRWST and into the sparger heads. The steam, in the form of high pressure jets, is injected into the IRWST where it is condensed and mixed with the IRWST water.

6.8.2.2.3 **Holdup Volume Tank (HVT)**

The HVT is a semi-circular tank located between the primary shield wall and the IRWST inner wall.

Any leakage from components or piping not routed to the RDT or spills inside containment, will drain to the HVT. This prevents direct contamination of the water in the IRWST.

A sump in the HVT collects any leakage. Pumps located in this sump transfer water to the Liquid Waste Management System (LWMS) during normal operation.

Question 410.102
(5.4.11)

b. Indicate whether the steam relief system (SRS) is designed to standards and codes that would be in conformance with Reg Guide 1.26, as related to the quality group classification of the piping system, and Reg Guide 1.29 position C.2, as related to the seismic design qualification of the system.

Response 410.102

b. The SRS is designed to standards and codes in conformance with RG's 1.26 and 1.29, position C.2. Further discussion of the Regulatory Guides can be found in Sections 3.2.1 and 3.2.2. This response will be added to Section 6.8.2.2.2 in a future amendment.

RAI 410.102(6)

Insert A

The steam relief system (SRS) is designed to standards and codes in conformance with Regulatory Guides 1.26 and 1.29, position C.2. Fur'her discussion on the application of RG's 1.26 and 1.29 are in Sections 3.2.1 and 3.2.2.

Question 410.103

a. Section 9.1.1.1 states compliance with the "intent" of Regulatory Guide 1.13 as a design basis. Considering that Regulatory Guide 1.13 pertains to spent fuel storage, explain what parts of the Guide, and to what extent, are met by the new fuel storage design.

b. Section 9.1.1.3.3 states that "new fuel storage racks and facilities are qualified as Seismic Category I." Identify the "facilities" which are so qualified.

c. Section 9.1.1.2 does not provide sufficient descriptive information on features illustrated in the figures. For instance, what is the function of "L" insert slots and boxes? How are the "cell blockers" attached to the structure? What is the equipment in the "new fuel inspection area"? What is their seismic classification?

d. The new fuel storage capacity changed from 166 in Amendment E to 121 in Amendment I. What is the design basis for the storage capacity of the system?

e. According to SRP Section 9.1.1, the design of the new fuel storage facility is acceptable if the integrated design is in accordance with, among other criteria, General Design Criteria 61 and 62 of 10 CFR 50, Appendix A. Specific criteria necessary to meet the requirements of GDC 61 and 62 are ANS 57.1 and ANS 57.3 as they relate to the prevention of criticality and to the aspects of radiological design. Provide information on the extent of compliance of the design to ANS 57.1 and ANS 57.3.

f. According to SRP Section 9.1.1, design calculations should show that the storage racks and the anchorages can withstand the maximum uplift forces available from the lifting devices without an increase in k_{eff}. A statement in the Safety Analysis that excessive forces cannot be applied due to the design is acceptable if justification is provided.

g. It is the position of the Plant Systems Branch that the vaults and racks of the new fuel storage facility are to be designed to preclude damage from dropped heavy objects. Provide the design features included in the design which either preclude the fall of heavy objects onto the racks or preclude damage from a drop of the load with the maximum potential energy.

h. Reference to Section 9.1.1.3.1.2.D in Section 9.1.1.3.1.1, regarding potential moderators such as fire extinguishing aerosols, appears to be in error. Should it be 9.1.1.3.1.2.C?

i. According to SRP Section 9.1.1, the failure of non-seismic Category I systems or structures located in the vicinity of the new fuel storage racks should not cause an increase in k_{eff} beyond the maximum allowable. Provide analysis that this condition is met or include in your application a commitment to the above condition as a design criterion.

Response 410.103

a. Although Regulatory Guide 1.13 pertains to the design of spent fuel storage racks, it is also used for the design of the new fuel racks. The applicable portions of the Regulatory Guide that are met are defined by Paragraphs 9.1.1.1.A. and 9.1.1.1.C.

b. The "facilities" associated with new fuel storage consist of the storage vault and the rack restraint system. The seismic category of other building components associated with handling fuel assemblies is noted in Table 3.2-1. (see response to NRC RAI 210.1)

c. The L-insert slots are provided in the wall of the fuel rack cavity (box) to permit the L-insert to be locked to the fuel cavity by its locking tab after it has been installed. The design of the locking tab and slot is such that the L-inserts can be remotely removed from the fuel racks, if required.

The cell blockers are installed in the fuel racks before the fuel assemblies are placed in the fuel rack and before the pool is flooded. The design is basically two concentric tubes with end restraints that limit the engagement of the tubes in the rack cavity wall (to avoid protrusion into an adjacent fuel rack cavity). The tubes are collapsed, installed into the fuel rack cavity, expanded into the holes in the fuel rack cavity wall, then locked together with a captured pin. In this manner the cell blockers are positively locked to the fuel racks but can be remotely removed if desired.

The new fuel inspection area is provided for the inspection of new fuel assemblies after they have been removed from their shipping container and before they have been placed in the fuel racks. It will contain a Seismic Category II inspection device to ascertain if the fuel assemblies meet the dimensional requirements for installation into the reactor vessel. Visual inspections will also be performed to check for shipping damage and to ensure that all protective wrapping material has been removed.

d. The number of new fuel assemblies required for a 12 month refueling cycle, an 18 month refueling cycle, and a 24 month refueling cycle was evaluated. This evaluation disclosed that a 24 month cycle is controlling from the standpoint of the maximum number of new fuel assemblies required, i.e., 108. Since the rack structure is square, the minimum array to accommodate 108 fuel assemblies at a density of 50% is two 11 x 11 fuel rack modules or 121 fuel assemblies.

e. The fuel handling equipment located in the new fuel storage area meets the requirements of ANS 57.1. The new fuel racks meet the requirements of ANS 57.3.

f. The lifting capacity of the overhead crane that is used to remove new fuel assemblies from the new fuel rack is restricted by either adjusting the motor stall torque or using load limiting devices. (See Paragraphs 9.1.1.3.1.1.D and 9.1.4.2.1.7.B).

g. The new fuel racks are located at the opposite end of the fuel building from the spent fuel pool to eliminate the possibility of moving heavy loads near the new fuel storage area. (See response to Question i). Administrative controls will be in place to limit the size of the load that can be carried over the new fuel racks so that the design impact energy that the racks can absorb without affecting k_{eff} will not be exceeded.

h. The reference section of Section 9.1.1.3.1.1 that discusses potential moderators should be 9.1.1.3.1.2.B instead of 9.1.1.3.1.2.D. This change will be incorporated in the next submittal.

i. The new fuel storage racks are located in a concrete vault at the opposite end of the fuel building from the spent fuel pool area to preclude passage of the spent fuel shipping cask overhead crane over the racks during the handling operations associated with spent fuel inspection, handling, and shipping. This location minimizes the number of systems or structures located in the vicinity of the new fuel storage facility. All systems or structures in the vicinity will be designated as Seismic Category II to preclude their failure and entry into the new fuel storage area.

The new fuel storage racks will be designed to limit the rack k_{eff} to 0.98 based on the postulated accident conditions and assumptions of Paragraph 9.1.1.3.1. No load will be permitted to be carried over the loaded fuel racks whose impact energy, if dropped, will exceed the impact energy of the postulated dropped fuel handling tool or the combination of the dropped fuel handling tool and fuel assembly. The maximum impact energy shall be limited such that dropped loads do not change the K_{eff} of the fuel array to more than 0.98.

Question 410.104

a. According to SRP Section 9.1.2, the design of the spent fuel storage facility is acceptable if the integrated design is in accordance with General Design Criterion 2. Acceptance for meeting this criterion is based on conformance to position C.3 of Regulatory Guide 1.13, the applicable portions f Regulatory Guide 1.29, Regulatory Guide 1.117, nd ANS 57.2 Paragraphs 5.1.1, 5.1.3, 5.3.2, and 5.3.4. Discuss the spent fuel storage design with respect to these criteria. What is the meaning of Section 9.1.2.3.2 regarding the "intent" of Regulatory Guide 1.13?

b. In Section 9.1.2.3.3 provide a list of the facilities which are Seismic Category I.

c. According to SRP Section 9.1.2, design calculations should be provided to show that the spent fuel racks and any anchorage can withstand the maximum fuel equipment uplift forces without an increase in k_{eff} or a decrease in pool water inventory. A statement in the Safety Analysis that excessive forces cannot be applied due to the design of the fuel handling equipment is acceptable if justification is provided.

d. According to SRP Section 9.1.2, the design of the spent fuel storage facility is acceptable if the integrated design is in accordance with General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety functions. Acceptance for meeting this criterion is based on conformance with Paragraph 5.4 of ANS 57.2. Provide the design features which satisfy GDC 63 and discuss compliance with Paragraph 5.4 of ANS 57.2.

Response 410.104

a. The design of the spent fuel storage facility meets the intent of Regulatory Guide 1.13, Position C.3, the applicable portions of Regulatory Guide 1.29, Regulatory Guide 1.117, and ANS 57.2, Paragraphs 5.1.1, 5.1.3, 5.3.2, 5.3.4. "Complies with the intent of" means that the design "meets our interpretation of the design objectives" of the regulatory guide. Since regulatory guides provide "guidance" and are not considered requirements, we prefer not to use the phrase "complies with". "Compliance" is reserved for firm requirements.

   Some examples of how the design meets the intent of the reference documents are as follows:

The depth of the spent fuel pool is such that when the irradiated fuel assembly is being carried over the spent fuel racks by the spent fuel handling machine at its maximum lift height, there is sufficient water coverage to ensure that personnel on the spent fuel handling machine or on the operating floor around the pool are not exposed to radiation levels exceeding 2.5 MREM per hour.

Piping penetrations to the spent fuel pool are at least 10 feet above the top the fuel assemblies when the assemblies are seated in the spent fuel racks. The bottom of the gates that lead from the spent fuel pool to the fuel transfer system canal and the spent fuel shipping cask laydown area are above the top of the stored fuel assemblies. The spent fuel racks and the pool floor are designed to withstand the maximum impact energy of a dropped fuel handling tool or a dropped fuel assembly with its handling tool from the maximum lift height. Redundant low and high level water alarm systems in conjunction with the pool skimmer system minimizes the potential for overfilling the pool.

b. The spent fuel pool concrete structure, the spent fuel pool rack support system, and the spent fuel racks are Seismic Category I. See Table 3.2-1 for a tabulation of the designated seismic categories for the fuel building components.

c. The spent fuel handling machine replicates the refueling machine in accordance with Paragraph 9.1.4.2.2.8. Paragraph 9.1.4.2.1.1 defines the interlocks that are incorporated on the refueling machine/spent fuel handling machine to limit fuel assembly handling loads.

d. The design of the spent fuel storage facility meets the intent of Paragraph 5.4 of ANS 57.2. As an example, the facility incorporates monitoring systems to verify pool water temperature to ensure adequate fuel assembly cooling, radiation detectors to determine if radiation levels exceed predetermined setpoints and alarms to notify plant personnel of abnormal conditions.

Question 410.105:

Section 9.1.3.3.1, Availability and Reliability, of your submittal states that "a cooling train may be shut down for limited periods of time for maintenance or replacements of malfunctioning components." However, it does not provide sufficient information to determine the rate of pool heatup and, thus, the allowable unavailability of the system, for normal and abnormal conditions. Please provide this information.

Response 410.105:

Shutdown of one cooling train for maintenance only occurs if the second train is available for continued cooling. Section 9.1.3.3.1, Availability and Reliability states that "in the event of the failure of a pool cooling pump or loss of cooling to a heat exchanger the second cooling train provides backup capability, thus assuring continued cooling of the spent fuel pool." Therefore, during maintenance of one cooling train, the spent fuel pool is not without cooling, and it does not experience a heatup beyond the limits specified in Section 9.1.3.1.4, which addresses the maximum pool temperatures reached with single train cooling.

Section 9.1.3.1.4, System Capacity Bases states that "given a single active failure, the maximum temperatures for normal conditions or a full core offload are 140°F or 180°F respectively." This statement applies to the situation where a cooling train is shut down for maintenance.

These sections show that when a spent fuel pool cooling train is shut down for maintenance, a second cooling train provides continued cooling of the spent fuel pool. Also, the maximum pool heatup temperatures are provided for normal and abnormal conditions during this period.

410.106 (a):

Section 9.1.3.2.1 states that the spent fuel pool cooling system is "safety-related, Quality Group C". Indicate whether or not the system is designed to seismic category I requirements. If not, confirm that the following systems are designed to seismic category I requirements and are protected against tornadoes: the fuel pool makeup water system and its source; and the fuel pool building and its ventilation and filtration system. Confirm that the make-up, and ventilation and filtration systems can withstand a single failure. Also, confirm that the transient temperature used in evaluating combined loads on structures is the boiling temperature of water.

Response to 410.106 (a):

As stated in Section 3.2.2 of CESSAR-DC, all fluid systems important to safety are classified in accordance with ANSI/ANS Standard 51.1. For purposes of CESSAR-DC, Quality Group C of Regulatory Guide 1.26 is equivalent to Safety Class 3 of ANSI/ANS 51.1. Consequently, the spent fuel pool cooling system (the safety related portion of the Pool Cooling and Purification System (PCPS)), is designated Safety Class 3.

The selection of Seismic Category I structures, systems and components is in accordance with the guidance provided in Regulatory Guide 1.29. Section 3.2.1 of CESSAR-DC specifies that all Safety Class 3 components are designated Seismic Category I. Therefore, the spent fuel pool cooling system is Seismic Category I. Table 3.2-1, Sheet 7 of 17, lists the major components of the PCPS and their Seismic Category I designations. Safety classifications (and, by reference to Section 3.2.1, the seismic categorizations) for small components are shown on the PCPS P&ID, Figure 9.1-3.

The refueling pool purification subsystem of the PCPS, which does not perform a safety related function, is classified non-nuclear safety (NNS) in accordance with the ANSI/ANS Standard 51.1, and Regulatory Guides 1.26 and 1.29.

Question 410.106

b. It is the position of the Plant Systems Branch that the design must meet the requirements of 10 CFR 20.1(c) as it relates to radiation doses being kept as low as reasonably achievable (ALARA). In meeting this regulation Regulatory Guide 8.8, positions C.2.f(2) and C.2.f(3) are used as a basis for acceptance. Discuss the features of the design which satisfy the above positions.

Response 410.106

b. The design features discussed below contribute to maintaining the personnel exposure in the Fuel Building ALARA and ensure compliance with 10 CFR 20.1(c) and Regulatory Guide 8.8, positions C.2.f(2) and C.2.f(3), requirements. Additional information regarding System 80+™ ALARA design features is provided in the System 80+™ ALARA Guidelines Manual previously submitted.

The Pool Cooling and Purification System (PCPS) provides the capability to remove radioactive materials from the spent fuel pool water. The operation of the PCPS and the provision of adequate shielding permit unrestricted access of plant personnel to the spent fuel pool area by maintaining the radiation level ALARA (< 2.5 mrem/hr). The PCPS equipment includes demineralizers, filters and strainers which provide for the removal and retention of radioactive material, such as corrosion products. The PCPS's design features, discussed in section 9.1.3 of the CESSAR-DC, include:

1. Provisions for the transfer of spent filter cartridges and resins to the Solid Waste Management System (SWMS) for processing and disposal.

2. Routing of spent resin transfer lines through shielded pipe chases to the SWMS.

3. Provision of a flushing capability for spent resin transfer lines to prevent clogging, which would create hot spots and require additional maintenance.

Floor drains in the spent fuel pool area are provided to collect and route radioactive liquid to the Liquid Waste Management System (LWMS) for processing. The floor drain system's design features, discussed in Section 9.3 of the CESSAR-DC, include:

1. Floor drain piping will be sloped to permit gravity flow of liquids to the sumps.

Response 410.106 (Cont'd)

2. Floors will be sloped to facilitate the collection of leakage from equipment or spills.

3. U bends will be provided to prevent migration of noble gases between elevations.

4. Curbing will be provided to facilitate the collection of spills from equipment.

The Fuel Building Ventilation System provides for a controlled monitored release pathway for gaseous effluent from the Fuel Building, as well as environmental control for the operation of equipment. The Fuel Building Ventilation System consists of 2-100% capacity exhaust systems. Each system is provided with a radiation monitor upstream of the filter beds. The Fuel Building Ventilation System normally exhausts in the bypass mode, except during fuel movement. Upon detection of high radiation, the Fuel Building Ventilation System is automatically switched to the filtered mode and an alarm signal is received in the Control Room. The exhaust is then vented through the unit, where it is monitored prior to release to the environment. This mitigates the dose consequences to plant personnel and the general public due to a fuel handling accident. System 80+™ ventilation systems are designed to provide air flow from areas of lower potential activity to areas of higher potential activity. This minimizes the potential for the spread of airborne contamination. Areas that have a high potential for contamination, such as Fuel Storage Areas, are kept at a slightly negative pressure to prevent the spread of airborne contamination to noncontaminated areas of the plant.

Question 410.107

a - Evaluate the structural design features of the refueling cavity water seal that would preclude a leak or failure from occurring. Include the possibility of a fuel assembly or other structure dropping on the seal.

b - If a seal failure/leak occurred, determine the time to lower a fuel assembly below the reactor vessel flange level before unacceptable dose rates from a lowered water level above spent fuel in the reactor core.

c - For a postulated seal failure/leak, evaluate containment dose rates from a lowered level above spent fuel in reactor core.

d - For a postulated seal failure/leak, evaluate the following parameters: makeup capacity, emergency procedures, fully loaded spent fuel pool thermal-hydraulic and dose effects including dose rate to someone trying to manually close the transfer tube valve to hydraulically isolate the spent fuel pool from the leak, time to cladding damage without operator action. Specifically provide the maximum allowable time to isolate the spent fuel pool from the transfer tube and refueling pool before there are unacceptably high dose rates in the spent fuel pool area and inadequate spent fuel pool cooling due to the level dropping below the minimum NPSH requirement above the elevation of the pool cooling suction inlet piping.

Response 410.107

a. The refueling pool seal is designed to be installed in one piece between the reactor vessel flange and the pool floor. All fabrication welds will be liquid penetrant inspected prior to installation to ensure adequacy. After the seal assembly has been set in place, it will be permanently attached to the reactor vessel flange and to an embedment plate in the pool floor. Penetrations in the seal plate for ventilation and access to the ex-core instrumentation will be covered by bolted access hatches equipped with double seals when the pool is flooded. The annulus between the seals will be pressure tested after the hatches have been installed to determine the sealing adequacy.

The pool seal is designed to withstand OBE displacements without leakage. The pool seal is designed to limit potential leakage resulting from SSE displacements. Pool seal inspection will be required as part of the post seismic recovery procedure. The pool seal is also designed to accommodate, without leakage, relative displacements between the pool floor and the reactor vessel due to normal plant operation.

During refueling operations with the pool flooded, the heavy lift components that pass over the pool seal are the reactor vessel head, the upper guide structure with its lift rig, and the upper guide structure lift rig. Administrative controls will require that prior to transfer of heavy loads over the pool seal, the fuel transfer tube valve or the gate between the fuel building transfer system canal and the spent fuel pool shall be closed. This is done to preclude any change to the spent fuel pool water level during a postulated heavy load drop on the pool seal which may result in containment pool draindown. In addition, administrative controls preclude the movement of heavy loads over the pool seal if the refueling machine contains a fuel assembly. The refueling machine is designated seismic category II so that it will not fall on the pool seal during seismic accelerations. The maximum clearance between the bottom of the refueling machine and the top of the pool seal is less than two inches to minimize the impact energy for the postulated accident condition of a dropped fuel assembly on the pool seal. The pool seal has been designed so that it will not leak as a result of this impact load.

b. It has been determined that a 24 square inch opening in the pool seal will result in pool draindown to the reactor vessel flange level in approximately 4 hours without additional water being added to the pool. It has also been determined that present plant systems are capable of maintaining the pool water level in the event there is a 24 square inch opening in the pool seal. A fuel assembly can be lowered below the reactor vessel flange level from the fully withdrawn position within 3 minutes.

c. With the water at the reactor vessel flange level, the radiation level at the pool seal area as a result of the fuel assemblies within the core will not be significantly greater than that with the reactor vessel head in place. The exposed CEA extension shafts will result in an increase in the overall radiation level. However, if it is necessary to do maintenance on the reactor vessel pool seal, temporary shielding can be placed around the extension shafts to reduce the radiation levels in the work area.

d. The responses to parts a, b, and c of this RAI address the concerns of this question.

Question 410.108

Provide numerical values of dose rate at appropriate locations above and around the spent fuel pool with its design maximum loading. Has the effect of higher than anticipated fuel enrichment and burnup been incorporated in the spent fuel pool shielding design?

Response 410.108

The radiation level in the work area around the spent fuel pool and on the spent fuel handling machine during fuel handling operations will not exceed 2.5 MREM per hour. The spent fuel pool shielding has been designed for the maximum anticipated enrichment and burnup of the fuel assemblies.

Question 410.109

a. It is the position of the Plant Systems Branch that the design for both light and heavy fuel handling load systems must conform to the requirements of General Design Criterion 2, as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes. The acceptance is based on meeting Regulatory Guide 1.29, position C.1 and C.2, and positions C.1 and C.6 of Regulatory Guide 1.13. In the safety evaluation area of the section specifically address the conformance of the design to the above guidelines.

b. It is the position of the Plant Systems Branch that the design of the fuel/load handling systems must conform to the requirements of General Design Criterion 61 as it relates to a radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure. Acceptance is based on the guidelines of positions C.3 and C.5 of Regulatory Guide 1.13, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, and guidelines in NUREG-0554, and NUREG-0612. In the safety evaluation area of the section, specifically address the conformance of the design features to each of the above guidelines.

Response 410.109

a. Conformance to the guidelines identified are addressed by the following inserts which will be included in a future revision to CESSAR-DC:

   1. The following will be added to Paragraph 9.1.4.3.1 of CESSAR-DC: " The fuel building overhead crane and the containment polar crane meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the cranes to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the cranes meet the intent of Regulatory Guide 1.13, Positions C.1 and C.2, and 57.1/ANSI-N208,ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612."

   2. The following will be added to paragraph 9.1.4.3.2 of CESSAR-DC: "The refueling machine and the spent fuel handling machine meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the equipment to withstand the effects of earthquakes. With

respect to radioactive release as a result of fuel damage, the machines meet the intent of Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS57.2/ANSI-N210, NUREG-0554, and NUREG-0612."

3. The following will be added to Paragraph 9.1.4.3.3 of CESSAR-DC: "The reactor vessel closure head lift rig and the reactor vessel internals lift rigs meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the lift rigs to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the lift rigs meet the intent of Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612.

b. See response to question 4.109a.

are required, they may be introduced into the upper guide structure at this time. The expended CEAs are moved to the CEA elevator, adjacent to the upper guide structure storage area, where the upper CEA casting is removed from the CEA rods utilizing special tooling. Each rod is picked up individually and placed into the transport container where the lower 15-foot section is cut off utilizing the portable underwater hydraulic CEA cutter. The upper 5-foot section of the CEA rod is then placed into the transport container and the operation is repeated until all rods have been cut. The transport container is then moved to the transfer carriage where it is transported to the spent fuel building for CEA rod disposal. E

At the completion of the refueling operation, the fuel transfer tube valve is closed. The upper guide structure is reinserted in the reactor vessel, the CEDM extension shaft assemblies and CEAs are lowered into position, and the lift rig is removed. The water in the refueling pool is lowered to the top of the extension shafts. The reactor vessel head is then lowered until the CEDM extension shaft assemblies are engaged by the control element drive mechanism nozzle funnels. Lowering of the head and the water level is continued until the head is seated. The remainder of the refueling pool water is then removed. Then the studs are installed, the head is bolted down, and the transfer tube penetration sleeve is sealed. The ICIs are reinserted into the core region and reconnected to their cabling. I

The head area cable tray is replaced, CEDM and HJTC cabling is connected, cooling ducts are reconnected to the CEDM cooling manifold, and the vessel vent piping is installed.

9.1.4.3 Safety Evaluation

9.1.4.3.1 Fuel Building Overhead Cranes and Containment Polar Crane

The containment polar crane, the cask handling hoist, and the fuel handling hoist are designed to prevent the drop of a heavy load such as the reactor vessel head or the spent fuel shipping cask. In addition, predetermined load paths for major lifts (see Figures 9.1-19 and 9.1-20), operator training, and regular crane maintenance minimize the possibility of load mishandling.

Limit switches, electrical interlocks and mechanical interlocks prevent improper crane operations which might result in a fuel handling accident. This is also discussed in Section 9.1.4.2.1.7. The spent fuel cask handling hoist is restricted from movement over the new and spent fuel storage areas when the

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fuel racks contain fuel assemblies. The new fuel handling hoist is restricted from movement over the spent fuel storage area when the spent fuel racks contain fuel assemblies.

In accordance with the regulatory position of Regulatory Guide 1.13 and General Design Criterion 61 of Appendix A to 10 CFR 50, the hoists are also restricted from passing over the spent fuel pool cooling system for ESF systems which could be damaged by dropping the load. I

Set points for the hoist interlocks are set to prevent falling or tipping of the loads into the fuel storage areas.

Administrative controls preclude movement of heavy loads within the containment building pool when the refueling machine contains a fuel assembly. During heavy load movement, the fuel transfer tube valve is closed to avoid water level changes in the fuel building during postulated accident conditions such as dropping the heavy load on the reactor vessel pool seal.

Insert 1

9.1.4.3.2 **Fuel Handling**

A failure modes and effects analysis is described in Table 9.1-2.

Direct voice communication between the control room and the refueling machine console is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Operability of the fuel handling equipment including the bridge and trolley, the lifting mechanisms, the upending machines, the transfer carriage, and the associated instrumentation and controls is assured through the implementation of preoperational tests and routines. Prior to the first actual fuel loading, the equipment is cycled through its operations using a dummy fuel assembly. In addition to the interlocks described in Section 9.1.4.2.1, the equipment has the following special features: E

A. The major systems of the fuel handling system are electrically interlocked with each other to assist the operator in properly conducting the fuel handling operation. Failure of any of these interlocks in the event of operator error will not result in damage to more than one fuel assembly.

B. Miscellaneous special design features which facilitate handling operations include:

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1. Backup hand operation of the refueling machine hoist and drives and CEA change platform traverse drives in the event of power failure.

2. Dual wound transfer system motor to permit applying an increased pull on the transfer carriage in the event it becomes stuck.

3. Viewing port in the refueling machine trolley deck to provide visual access to the reactor for the operator.

4. Electronic and visual indication of the refueling machine position over the core.

5. Protective shroud into which the fuel assembly is drawn by the refueling machine.

6. Manual operation of transfer system upenders by a special tool in the event that the hydraulic system becomes inoperative. E

7. Removal of the transfer system components from the refueling pool for servicing without draining the water from the pool.

C. The fuel transfer tube is sufficiently large to provide natural circulation cooling of a fuel assembly in the unlikely event that the transfer carriage should be stopped in the tube. The manual operator for the fuel transfer tube valve extends from the valve to the operating deck. Also, the valve operator has enough flexibility to allow for operation of the valve even with thermal expansion of the fuel transfer tube.

D. Mechanical stops in both the refueling and spent fuel handling machines restrict withdrawal of the spent fuel assemblies. This results in the maintenance of a minimum water cover of 9 feet over the active portion of the fuel assembly. The resulting radiation level from the spent fuel is 2.5 mrem/hr or less in the work area when the shielding of the fuel handling machine is taken into account.

Insert 2.

E

9.1.4.3.3 Reactor Vessel Closure Head Handling

The reactor vessel closure head lift rig is designed, tested, and inspected to meet the intent of NUREG-0612 and the design criteria of ANSI N14.6. Analyses for the postulated head drops is performed to assure that the reactor vessel support system and

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shutdown cooling supply flow paths remain functional, that the core will remain in a coolable configuration, and that the k_{eff} of the core will remain below 0.95. E

Insert 3.

9.1.4.4 Testing and Inspection Requirements

During manufacture of the fuel and CEA Handling Equipment at the vendor's plant, various in-process inspections and checks are required including certification of materials and heat treating, and liquid-penetrant or magnetic-particle inspection of critical welds. Following completion of manufacture, compliance with design and specification requirements is determined by assembling and testing the equipment in the vendor's shop. Utilizing a dummy fuel assembly having the same weight, center of gravity, exterior size and end geometry as an actual assembly, all equipment is run through several complete operational cycles. In addition, the equipment is checked for its ability to perform under the maximum limits of load, fuel mislocation and misalignment. All traversing mechanisms are tested for speed and positioning accuracy. All hoisting equipment is tested for vertical functions and controls, rotation, and load misalignment.

Hoisting equipment is also tested to 125% of specified hoist capacity. Setpoints are determined and adjusted and the adjustment limits are verified. Equipment interlock function, and backup systems operations are checked. Those functions having manual operation capability are exercised manually. During these tests, the various operating parameters such as motor speed, voltage, and current, hydraulic system pressures and load measuring accuracy and setpoints are recorded. At the completion of these tests the equipment is checked for cleanliness, and the locking of fasteners by lockwire or other means is verified.

Equipment installation and testing at the plant site are controlled by approved installation procedures and preoperational test procedures designed to verify conformance with procurement specifications. Each component is inspected and cleaned prior to installation into the system. Recommended maintenance, including any necessary adjustments and calibration, is performed prior to equipment operation. Preoperational tests also include checks of all control circuits including interlocks and alarm functions.

The following testing and inspections will be used for both the containment polar crane and fuel building overhead crane. E

A. Hoists and cable will be load tested at 125% of the rated load.

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Insert 1.

" The fuel building overhead crane and the containment polar crane meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the cranes to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the cranes meet the intent of Regulatory Guide 1.13, Positions C.1 and C.2, and 57.1/ANSI-N208,ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612."

Insert 2.

"The refueling machine and the spent fuel handling machine meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the equipment to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the machines meet the intent of Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS57.2/ANSI-N210, NUREG-0554, and NUREG-0612."

Insert 3.

"The reactor vessel closure head lift rig and the reactor vessel internals lift rigs meet the intent of Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the lift rigs to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the lift rigs meet the intent of Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612.

Question 410.120
(9.4.6)

The CESSAR states, on pg. 9.4-27, that the containment high volume purge mitigates the consequences of fuel handling accidents to within 10CFR100 limits and, on pg. 9.4-30, that the system is not an ESF. Reconcile the apparent contradiction.

Response 410.120
(9.4.6)

The Containment High Volume Purge System is not an Engineered-Safety-Feature system. During a postulated fuel handling accident, the charcoal filtration is credited with filtration of the release, but no credit is allowed for release reduction resulting from containment isolation or mixing in the containment atmosphere prior to the release.

A mark-up of CESSAR-DC Sections 9.4.6.1, 9.4.6.3 and 16.12.3 are attached for NRC review and will be incorporated in the next amendment.

B. Entering and leaving chilled water temperatures at supply ventilation units.

C. Air flow rates for the supply and exhaust units.

D. Chilled water flow rates to supply ventilation units.

9.4.6 CONTAINMENT COOLING AND VENTILATION SYSTEM

9.4.6.1 Design Basis

The Containment Cooling and Ventilation System is designed to maintain acceptable temperature limits inside containment to ensure proper operation of equipment and controls during normal plant operation, normal shutdown and for personnel access during inspection, testing, and maintenance. It is comprised of the following subsystems as shown on Figure 9.4-6.

E

I

A. The containment recirculation cooling subsystem functions during normal plant operation to maintain a suitable ambient temperature for equipment located within the containment. This system also operates during a loss of offsite power.

B. The control element drive mechanism (CEDM) cooling subsystem functions during normal plant operation to maintain a suitable air temperature around the rod drive mechanisms.

E

C. The containment air cleanup subsystem operates before and during personnel entries to reduce airborne radioactivity.

D. The cavity cooling subsystem functions to maintain a suitable air temperature in closed ended cavities.

E. The high purge supply operates before and during personnel entries to reduce airborne radioactivity.

F. The low purge is a pressure relief system used to relieve containment pressure during start-up or shutdown. The IRWST purge supply and exhaust is normally closed. It is opened only for personnel access.

I

G. The containment high volume purge mitigates the radiological consequences of a postulated fuel handling accident inside containment. Dose at site boundary is well within the guidelines of 10 CFR 100.

Insert 1 →

The containment recirculation cooling subsystem is designed to maintain the average containment air temperature between 60°F and 110°F during normal plant operation with three of four cooling

Q410.120

Insert 1

The Containment High Volume Purge System is not an Engineered-Safety-Feature system. During a postulated fuel handling accident, the charcoal filtration is credited with filtration of the release, but no credit is allowed for release reduction resulting from containment isolation or mixing in the containment atmosphere prior to the release.

The CEDM cooling system consists of two 100% capacity cooling units each with cooling coils and fan for heat removal with a continuous flow of cool air across the drive mechanisms where it is released into the containment and returned over the bridge wall to be recirculated.

The low purge has two 100% capacity supply fans and two 100% capacity exhaust fans with filters and heat coil to temper the supply air and a filter train for the exhaust.

The high purge has two 100% capacity supply fans and two 100% capacity exhaust fans with filters, heating coil, cooling coil to temper the supply air and a filter train for the exhaust. I

The containment air cleanup system consists of two filtration units, each with primary filter, HEPA filter, carbon filter, and centrifugal fan. The units circulate a portion of the containment atmosphere for cleanup prior to and during a personal entry into containment. They also serve to reduce airborne activity prior to making a routine atmospheric release of containment air.

E

The containment recirculation cooling system consists of four 33% capacity recirculation cooling units, each connected to and associated recirculation fan.

The CEDM cooling system consists of two 100% capacity cooling units, each with associated 100% capacity fan.

The containment air cleanup systems each have one third capacity and the high purge has one third capacity to give a total of 100% capacity required by ANSI/ANS-56.6. I

The cavity cooling subsystem consists of two 100% capacity supply fans.

9.4.6.3 Safety Evaluation

The Containment Cooling and Ventilation System provides adequate capacity to assure that proper temperature levels are maintained in the containment under operating conditions. Sufficient redundancy is included to assure proper operation of the system with one active component out of service. Although not required, this system operates to maintain the containment temperature within acceptable limits during a loss of offsite power. E

The Containment Cooling and Ventilation System is not an Engineered Safety Feature and no credit has been taken for the operation of any subsystem or component in analyzing the consequences of design basis accidents. The High Volume Purge System charcoal filtration is credited with filtration of the release from a postulated fuel handling accident.

16.12.3 3.9.3 CONTAINMENT PENETRATIONS

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment building penetrations shall be in the following status:

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

b. One door in each airlock closed,

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

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

2. Capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

Exhausting through operable Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers, and is

APPLICABILITY: During CORE ALTERATIONS
During movement of irradiated fuel within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.1 One or more containment penetrations not in the required status.	A.1 Suspend CORE ALTERATIONS	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel within containment.	Immediately

Question 480.35a:

Although Section 6.5.1.3.K.3 of the CESSAR indicates that a calculation for NPSH for the containment spray pumps was performed, there is insufficient information available. As required by GDC 38 and SRP 6.2.2 Rev. 4, Item II.2, provide an NPSH analysis for the containment spray pumps to ensure that pump cavitation will not occur during any anticipated operating conditions. This analysis should be performed in accordance with Regulatory Guide 1.82, Rev. 1 and Regulatory Guide 1.1, for both the injection and recirculation phases of a postulated LOCA. The analysis should be based on maximum expected temperature of the pumped fluid and no increase in containment pressure from that present prior to postulated LOCAs. This analysis should have sufficient detail to permit the staff to determine the adequacy of the analysis.

Response 480.35a:

The Containment Spray pumps for System 80+ take suction from the In-containment Refueling Water Storage Tank (IRWST). Because this tank is inside containment, there is no distinction between the injection and recirculation modes of operation.

The available net positive suction head (NPSHA) was determined using the following equation:

$$NPSHA = H_a + H_{ss} - H_f - H_{vp}$$

Where: H_a is the pressure (in feet of fluid) that exists at the free surface of the suction source (IRWST).

H_{ss} is the elevation head from the pump suction to the surface of the suction source.

H_f is the head loss due to fluid flow.

H_{vp} is the vapor pressure (in feet of fluid) of the fluid at the pump section.

The NPSH for the CS pumps was calculated assuming that IRWST water temperature is equal to the worst-case containment temperature and that the containment pressure corresponding to this condition is equal to the vapor pressure for this temperature. These assumptions ensure that no credit is taken for containment pressure since the containment and vapor pressure terms cancel out of the NPSH equation. Moreover, these assumptions agree with the intent of Regulatory Guide 1.1.

The NPSHA equation then reduces to:

$$NPSHA = H_{ss} - H_f$$

The maximum expected flow rate for the CSS will result in the minimum NPSHA.

The calculated value for NPSHA for the Containment Spray pumps is approximately 21 feet. This includes both Safety Injection and Containment Spray pumps taking suction from the IRWST at runout flows. This NPSHA is sufficient for pumps commonly available for such service. Equipment has not been procured for System 80+. The procurement documentation will specify ample margin between the NPSH available and the NPSH required. Confirmation of these requirements will be done during the detailed engineering phase when vendor data and drawings are available.

Question 480.35(c)

As stated in SRP 6.5.2, Rev 01, Item II.1.a, the operating period of the containment spray system should not be less than 2 hours, and the system should be capable of operation in the recirculation mode, on demand, for a period of at least 1 month following the postulated accident. What is the design operating period of the containment spray system?

Response

Revision 2 of SRP 6.5.2 was issued in December of 1988. Revision 2 no longer provides a 1 month post-accident operating period as an acceptance criteria. Revision 2 does, however, state that the operating period should not be less than 2 hours.

The System 80+ CSS is designed to be consistent with the requirements of ANSI/ANS-56.5-1979 (Section 7.3.4) which states that "spray equipment shall be qualified for operation in its post-accident environment for a minimum of 30 days".

The design operating period for the System 80+ CSS will meet or exceed the design operating period for previously licensed System 80 Containment Spray Systems.

Question 480.35e

As stated in SRP 6.5.2 Rev. 1, Item II.1.g, the pH of the aqueous solution collected in the containment sump after completion of injection of containment spray and ECCS water, and all additives for reactivity control, fission product removal, or other purpose, should be maintained at a level sufficiently high to provide assurance that significant long-term iodine re-evolution does not occur. Long-term iodine retention with no significant re-evolution may be assumed only when the equilibrium sump pH, after mixing and dilution with the primary coolant and ECCS injection, is above 8.5. CESSAR Section 6.5.3 indicates that the long-term pH of the recirculated containment spray solution will be maintained at a minimum of 7.0. Justify the difference between the two long-term pH values.

Response

Revision 2 to Standard Review Plan (SRP) 6.5.2 was issued in December of 1988. Item II.1.g of Revision 2 states long-term iodine retention may be assumed only when the equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7. System 80+ complies with this design requirement.

QUESTION 730.9

a. There appears to be a discrepancy between the description of the flow control valve on each subtrain of the EFWS and the graphical representation in Figure 10.4.9-1. In one instance the valve is shown to be a motor operated valve (Figure 10.4.9-1) instead of an air or pneumatic flow control valve as shown in Appendix 10A. Clarify the discrepancy and if the valve is air operated identify the failed state of the valve and if the valve is air operated identify the failed state of the valve on a loss of air and describe how the air supply was included in the system reliability analysis of Appendix 10A.

b. In the reliability analysis of Appendix 10A different failure probabilities are used for the two motor-driven pumps failure to run, 1.918E-03 and 6.384E-04. Explain the difference.

c. Similarly explain the difference in the probabilities used for the failure of the distribution valves in EFW subtrain A1 and subtrain A2.

RESPONSE 730.9

a. Figure 10.4.9-1 reflects the current design. The System 80+ Probabilistic Risk Assessment (PRA) Analysis is currently being updated to reflect these design changes. The reliability analysis for the EFWS presented in Appendix 10A also will be revised.

b. The failure probability rate of 6.384E-04 should have been used for both motor-driven pumps failure to run. The error is made in the conservative direction since a higher failure probability rate has been used for one pump. Therefore, there is no adverse impact on the results.

c. As explained in the response in part "a," there have been many evolutions with regard to power assignments for the distribution valves. The reliability analysis presented in Appendix A is based on the following power assignment for the distribution valves:

  EFW control valves and SG isolation valves in EFW subtrains A1 and B1 powered from the 125 VDC buses.

  EFW control valves and SG isolation valves in EFW subtrains A2 and B2 powered from the 480 VAC motor control centers (MCCs).

A loss of power from the electrical distribution system is an element contributing to failure probability of the valve associated with that source. Since there are two different power sources for distribution valves in the EFW subtrains A1 and A2 (and subtrains B1 and B2), the failure probabilities would naturally be different.