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ASEA BROWN BOVERI

March 15, 1991
LD-91-013

Project No. 675

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

Subject: Response to NRC Requests for Additional Information

Reference: NRC Letter, Request for Additional Information, T. J. Kenyon (NRC) to A. E. Scherer (C-E), dated June 26, 1989

Dear Sirs:

The reference letter 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 and Enclosure II provides the corresponding revisions to CESSAR-DC.

Should you have any questions on the enclosed material, please contact me or Mr. S. E. Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for EHK

E. H. Kennedy
Manager
Nuclear Systems Licensing

EHK:lw

Enclosures: As Stated

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

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Enclosure I to
LD-91-013

RESPONSES TO REQUESTS

FOR

ADDITIONAL INFORMATION

Question 810.1

Chapter 13 of CESSAR-DC provides extensive description of design requirements for the Technical Support Center (TSC) and the Emergency Operations Facility (EOF). Emergency preparedness regulations and related guidelines contain requirements and guidance for facilities and functions in addition to the TSC and EOF.

For the additional facilities listed below, (1) provide a description of the pertinent design requirements or guidance, or (2) cite the location of these descriptions in current or projected design requirements, or (3) describe or identify how the equivalent function is contained in the design requirements of another facility (e.g., many OSC functions might be reflected in Control Room design):

a. Operations Support Center (OSC)

b. Laboratory Facilities (fixed or mobile)

c. Post Accident Sampling System

d. Onsite Decontamination Facility

Response 810.1

In response to the NRC request, descriptions have been developed for each of the facilities listed above. These descriptions are enclosed and will be included in CESSAR-DC in a future amendment.

Question 281.32

Section 6.5.1.3.K., Chemistry and Sampling, indicates that the containment spray system is designed for 2.5 w/o boric acid at a pH of > 7.0. Discuss the spray additive or pH control system and describe how it meets Standard Review Plan Section 6.5.2, Containment Spray As A Fission Produce Cleanup System and meets the requirements of GDC 41, 42 and 43.

Response 281.32

The information for this response is provided in CESSAR-DC Section 6.5.3, Amendment I. Information is also provided in the resolution to Generic Safety Issue C-10 of CESSAR-DC, Appendix A, Amendment I.

Question 281.33

Section 9.1.2.2.2, Spent Fuel Pool Storage Racks, indicates that the structural design of the spent fuel and pool includes provisions for neutron poison inserts to meet future expansion potential. Since this is a likely situation based on current experience, the spent fuel racks with neutron poison inserts should be considered in the reference design. For a spent fuel rack design that includes neutron poison inserts, a coupon surveillance program should be included to monitor the performance of the neutron poison material in the spent fuel pool environment.

Response 281.33

Although poison inserts could be used with the fuel racks described, poison inserts are not included in this design and no credit was taken for them in the analysis. Specification and NRC review of a coupon surveillance program would be the responsibility of the organization proposing to use poison inserts at some future time.

Question 281.34

Describe the instrumentation and sampling to monitor the water purity and need for demineralizer resin replacement including the chemical and radiochemical limits and demineralizer differential pressure used to initiate corrective action (Section 9.1.3.3.3).

Response 281.34

Spent fuel pool and demineralizer effluent will be monitored by grab samples with laboratory analysis. The fuel pool will be monitored to ensure that the water quality is maintained within the limits specified in Section 9.1.3.3.3 of CESSAR-DC.

Demineralizer replacement is to be based on three criteria:

1. Breakthrough of cesium, cobalt, chloride, or fluoride.

2. Pressure drop not to exceed demineralizer and resin vendors' recommended limit for the as-procured equipment.

3. Thermal excursion approaching resin vendors' recommended limit for the as-procured equipment.

Section 9.1.3.3 of CESSAR-DC will be revised to include the above response.

Question 281.57

Provide a technical analysis and evaluation of the containment spray system's effectiveness in reducing containment pressure and temperature and lowering radioisotope releases during postulated dominant severe accident sequences. Discuss specific system design features for enhancing the mitigation of severe accident consequences.

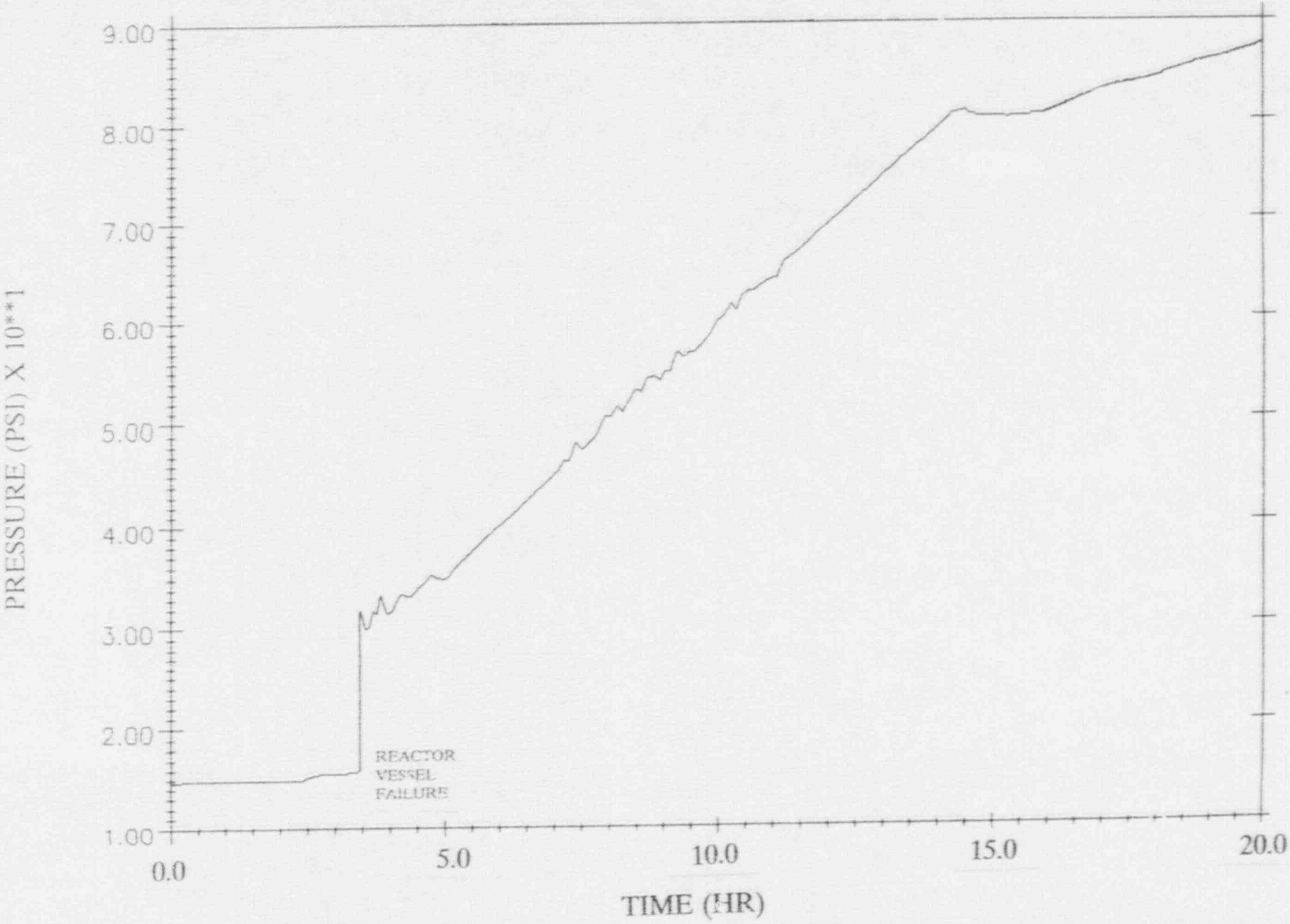
Response 281.57

An evaluation of the containment spray system is provided in Section 6.5.3 of Amendment I to CESSAR-DC. This section references Section 15.6.5 and Appendix 15A for a discussion of the effectiveness of removing elemental iodine from the containment atmosphere. The MAAP computer code is used for analysis of severe accident scenarios. The attached figures show the effect of the containment spray system in reducing the containment temperature and pressure for a severe accident. The severe accident analyzed is Station Blackout (SBO) and cases of normal spray initiation, delayed initiation, and no sprays are presented.

Design features which enhance the mitigation of severe accident consequences are described in Section 6.5. The safety-grade classification of the Containment Spray System (CSS) provides reasonable assurance that its mitigative function will be accomplished in severe accident environments similar to those predicted by MAAP. Comparison of the containment pressure and temperature for design basis events in Section 6.2.1 of CESSAR-DC to those for the attached case with "early" (normal) spray initiation indicates similar results for design basis and severe accidents. Please note, however, that the design of the CSS results from the design bases listed in Section 6.5, not from specific MAAP analyses.

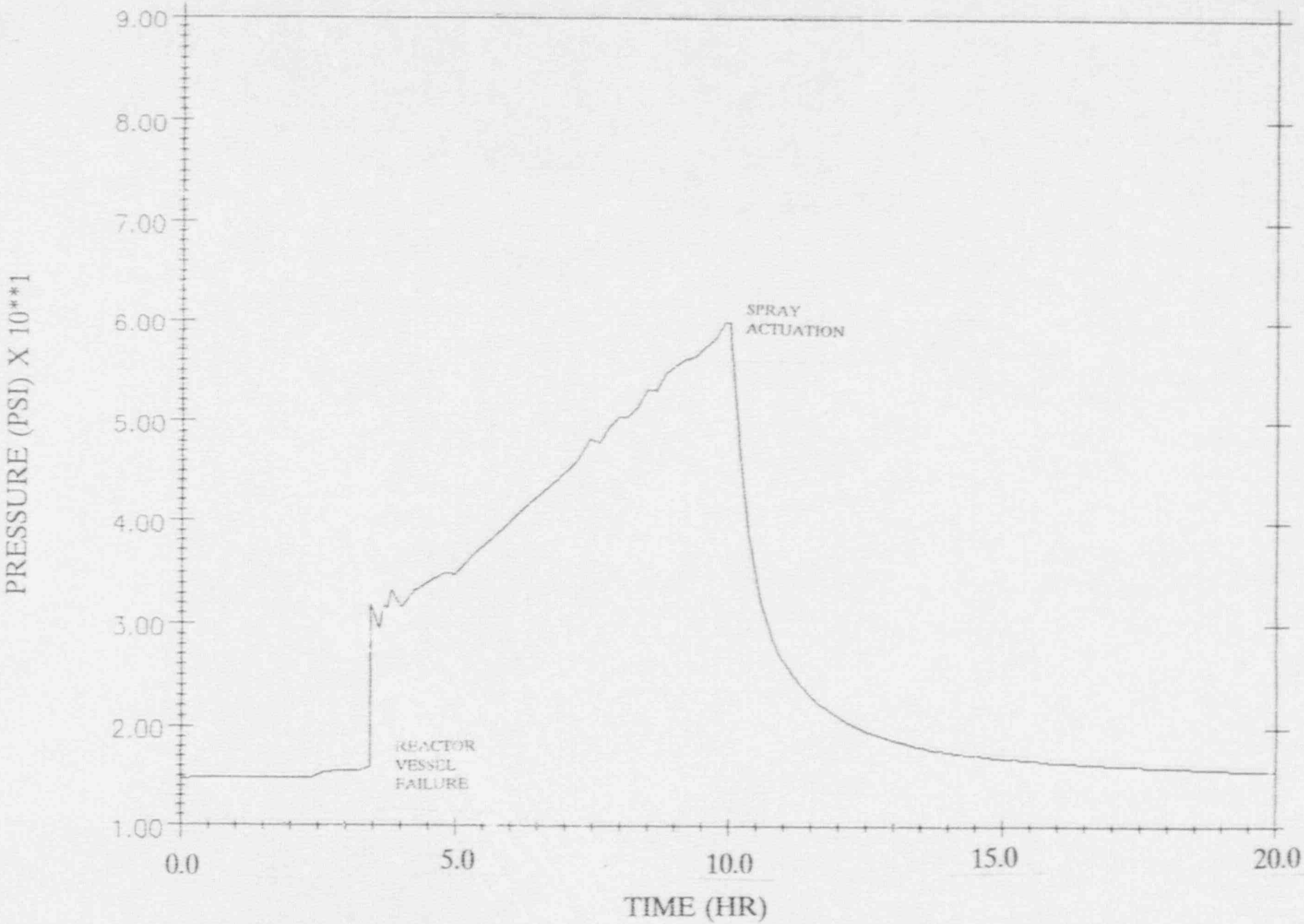
SBO W/O SPRAYS

CONTAINMENT PRESSURE VS. TIME



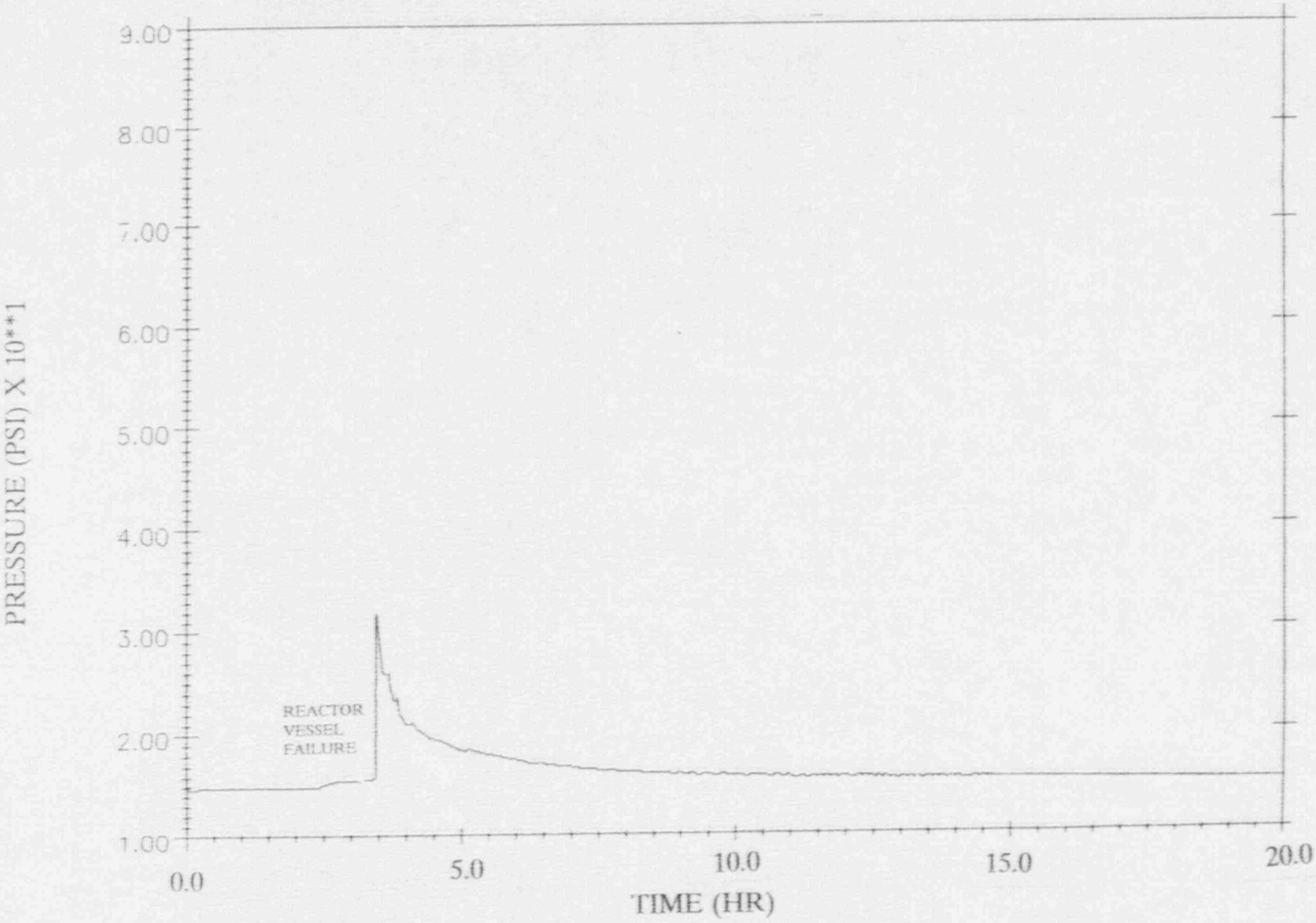
SBO W/LATE SPRAYS

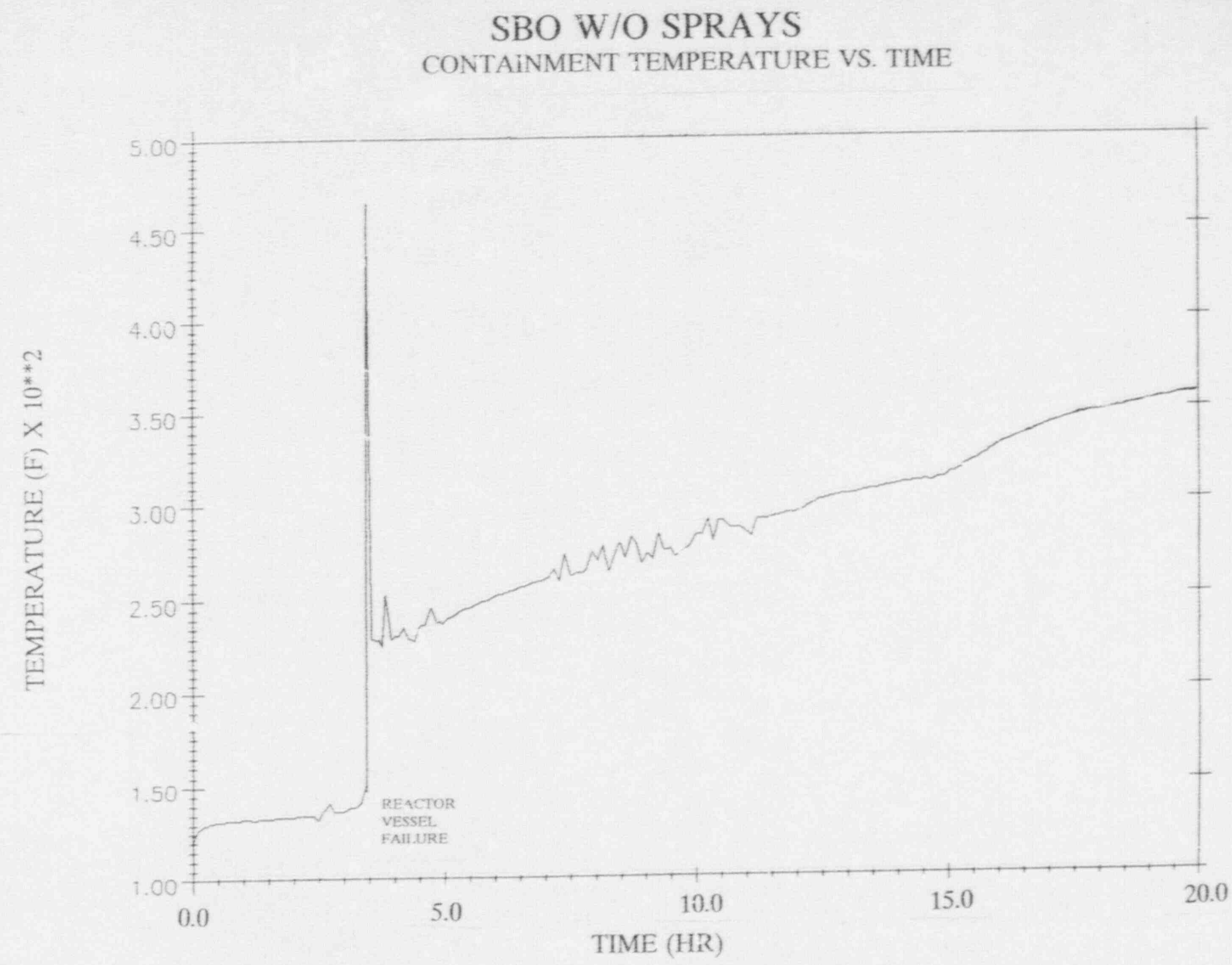
CONTAINMENT PRESSURE VS. TIME



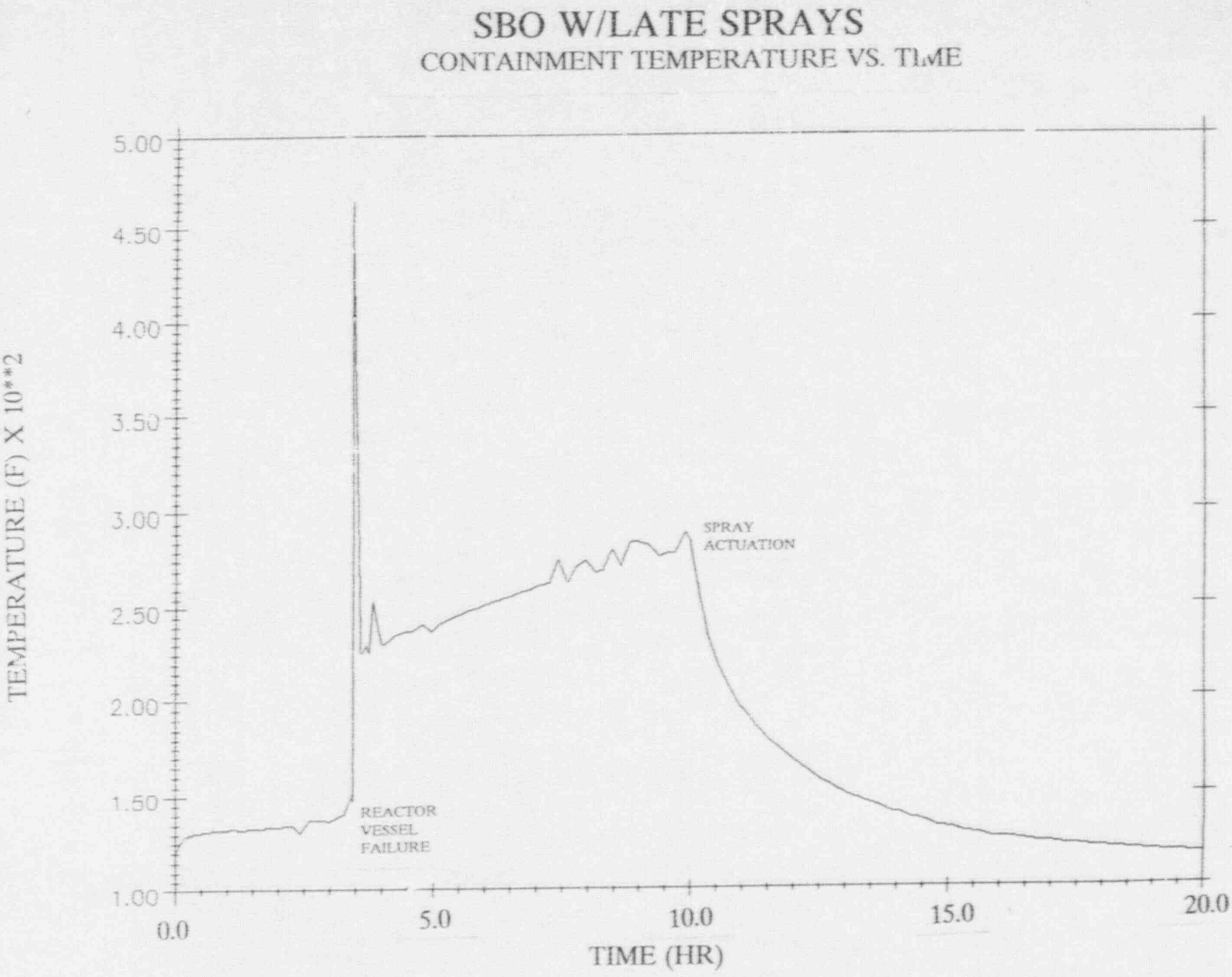
SBO W/EARLY SPRAYS

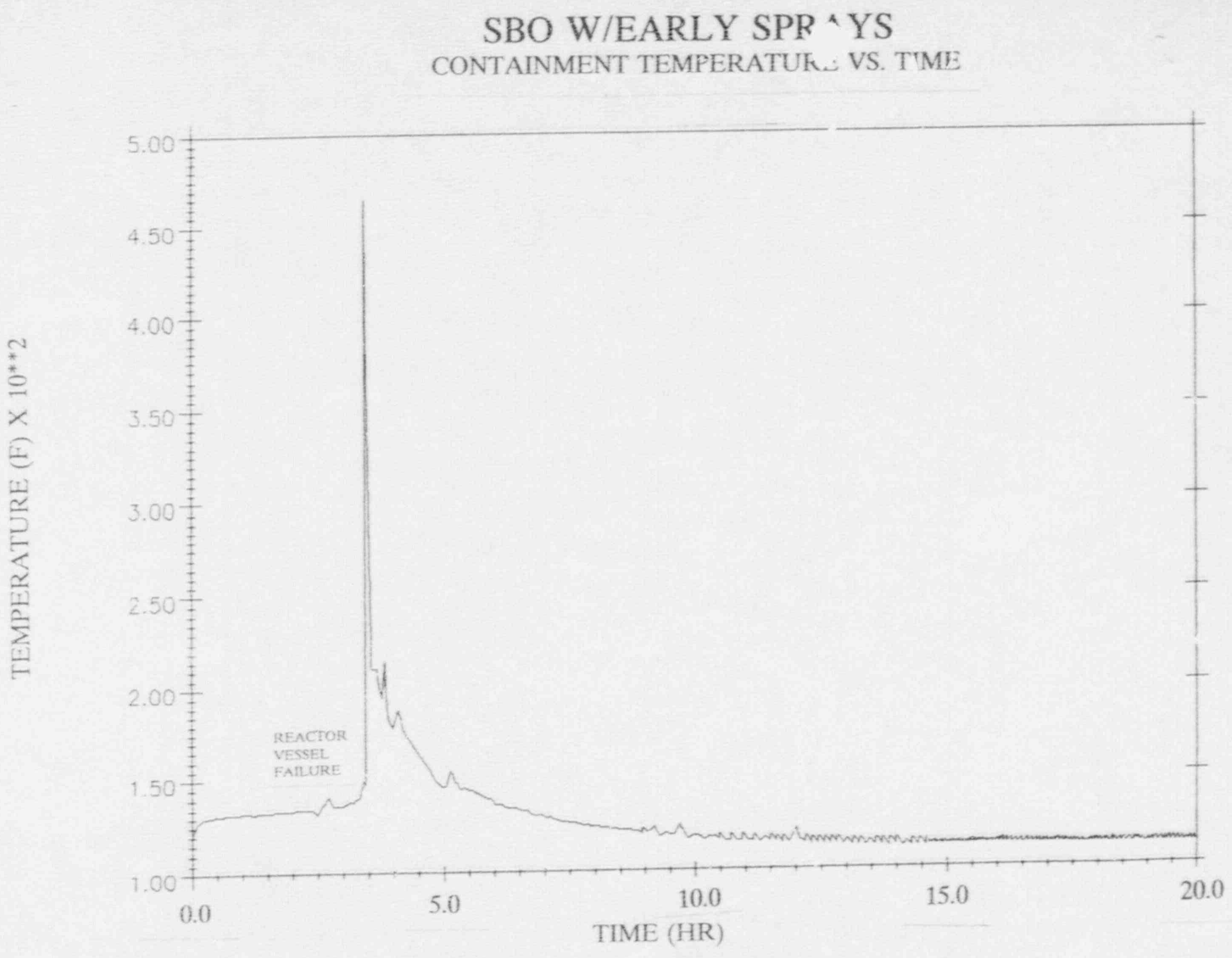
CONTAINMENT PRESSURE VS. TIME





SBO W/O SPRAYS
CONTAINMENT TEMPERATURE VS. TIME
TEMPERATURE (F) X 10**2
5.00
4.50
4.00
3.50
3.00
2.50
2.00
1.50
1.00
REACTOR
VESSEL
FAILURE
0.0
5.0
10.0
15.0
20.0
TIME (HR)


SBO W/LATE SPRAYS
CONTAINMENT TEMPERATURE VS. TIME
TEMPERATURE (F) X 10**2
5.00
4.50
4.00
3.50
3.00
2.50
2.00
1.50
1.00
0.0
5.0
10.0
15.0
20.0
TIME (HR)
SPRAY
ACTUATION
REACTOR
VESSEL
FAILURE


SBO W/EARLY SPR YS
CONTAINMENT TEMPERATU VS. TIME
TEMPERATURE (F) X 10**2
5.00
4.50
4.00
3.50
3.00
2.50
2.00
1.50
1.00
REACTOR
VESSEL
FAILURE
0.0
5.0
10.0
15.0
20.0
TIME (HR)

Question 281.58

Provide specific results of a failure modes and effects analysis (FMEA) of the containment spray system showing that the system is capable of withstanding a single failure without loss of function.

Response 281.58

Amendment E to CESSAR-DC contains in Table 6.5-3 the results of a failure modes and effects analysis for the Containment Spray System (CSS).

Question 281.59

Provide a detailed description and evaluation of all systems that interface with or support the containment spray system. This should include the potential for support/interfacing system single failures rendering the containment spray system inoperable and all potential systems interactions which could degrade plant safety.

Response 281.59

The systems which support the CSS are: 1) Incontainment Refueling Water Storage Tank (IRWST), 2) Component Cooling Water System (CCS), 3) Electrical System, 4) Engineered Safety Features Actuation System (ESFAS). The systems which passively interface the CSS are : 1) Shutdown Cooling System (SCS) and 2) Safety Injection System (SIS). The supporting and interfacing systems are evaluated as follows:

IRWST

The IRWST is discussed in Section 6.8. Each CSS has its own line feeding from the IRWST. A failure of one will not render the other CSS inoperable. The IRWST is composed of the lower regions of the containment lines with stainless steel. Water will be available to each CSS pump even if a rupture of the liner occurs. The IRWST is equipped with two independent safety grade level indicators and alarms.

CCS

The CCS is described in Section 9.2.2. The CSS is designed such that no single failure will prevent accomplishment of its safety function as stated in Section 9.2.2.1.1(c) Safety Design Bases, and further in Section 9.2.2.2 System Description (where independence and redundance is discussed).

Electrical System

Single failure redundancy of electrical power to the CSS pumps, valves and instrumentation is discussed in 6.4.1.3 A, E, F and I.

ESFAS

The ESFAS is discussed in Section 7.3. The system is designed to prevent a single failure from rendering the CSS inoperable.

SCS

There exist cross-over lines between each CSS and its companion SCS as shown on Figure 6.3.2-1A. Each cross-over line is equipped with a locked-closed, manual isolation valve. The purpose of the lines is for operational convenience, and they are not required for any active safety function. The CSS and SCS are both Safety Class 2 systems.

SIS

The CSS is connected to the SIS by small lines with locked-closed, manual valves which connect to a common return to the IRWST. They are not required for any active safety function.

Question 281.60

Provide the following information in order to permit the staff to perform an integrated review of the containment spray system (CSS):

a. Legible copies of the CSS Piping and Instrumentation Diagrams (P&IDs),

b. CSS heat exchanger fouling factors (design and expected values),

c. Design capacity of each CSS train, and

d. Technical data for CSS backup water source (outside containment) including source, transfer capability, pressure and flow data (see CESSAR-DC Section 6.3.2.2.1).

Response 281.60

a. Figures 6.3.2-1A and 6.3.2-1B of CESSAR-DC (Amendment I) include the CSS P&ID.

b. Fouling is not a problem for the CSS heat exchanger because of the highly purified and controlled water on both sides (IRWST water on tube side and CCS water on the tube side). The fouling factor assumed in design for tube and shell is 0.0005 hr-ft^2-°F/Btu. This is a conservative assumption; in service, the fouling factor is expected to be less.

c. The design capacity of each CSS train is:

CSS Flow	5000 gpm
CCW Flow	8000 gpm
Heat exchanger capacity	108 x 10^6 Btu/hr
CSS temp in	218°F
CSS temp out	175°F
CCW temp in	120°F
CCW temp out	147°F
Sizing condition	110% of decay heat at 24 hrs.

d. The CSS takes suction from the <u>in-containment</u> refueling water storage tank. Since the water source is inside containment, there is no need for an external, backup water source.

Question 281.61:

Provide the sprayed and unsprayed containment volumes and post-accident containment mixing features to ensure acceptable spray coverage of the entire containment per the guidance of SRP 6.5.2, Sections II.1.b and II.1.c.

Response 281.61:

The containment spray system is designed to provide coverage for 90% of the containment net free volume as recommended by SRP 6.5.2. This is the percent assumed in the iodine washout rate calculation. The remaining 10% of the containment net free volume is assumed to be unsprayed.

Question 410.47:

Provide a table listing the following parameters which are used to evaluate postulated piping failures in fluid systems:

(a) actual pipe dimensions
(b) system locations
(c) piping drawings
(d) design temperatures, and
(e) design pressures

Response 410.47:

Tables 3.6-3 and 3.6-4 of Amendment I identify system locations, piping drawings, temperatures, and pressures for high energy lines inside and outside containment. Consistent with the approach in Appendix C of Branch Technical Position SPLB 3-1 (SRP Section 3.6.1), emphasis is placed on location of piping and physical separation to minimize the effects of high energy line breaks. When specific plant components are procured, the need for special features to protect that equipment is evaluated and, if necessary, measures such as protective shields are taken.

Question 410.48:

Provide the following information (now shown as "LATER") in order to permit the staff to perform an integrated review of the postulated piping failures in fluid systems:

a) Completed CESSAR-DC Tables 3.6-3 concerning high energy lines within containment),

b) CESSAR-DC Section 3.5.4D concerning cross-reference sections for interface requirements on missile protection,

c) Completed Table 3.2-1 (sheet 4 of 6), classification of structures, systems, and components concerning the component cooling water system, spent fuel pool cooling and cleanup system, and station service water system, and

d) Completed Table 3.2-4, summary of criteria - structures.

Response 410.48:

a) Tables 3.6-3, "High Energy Lines Within Containment", and 3.6-4, "High Energy Lines Outside Containment", have been included in Amendment I.

b) Section 3.5.4D has been deleted in Amendment I.

c) Revisions to Table 3.2-1, "Classification of Structures, Systems, and Components", have been included in Amendment I.

d) Table 3.2-4, "Summary of Criteria - Category I Structures" has been included in Amendment I.

Question 410.49:

Clarify the criteria used for protection against the dynamic effects associated with postulated piping failures. Discuss how these criteria meet the guidance of BTP ASB 3-1, and GDC 4 which require the following:

a) adequate physical separation and remote location,
b) suitably designed protective enclosure, and
c) restraints and protective measures.

Response 410.49:

Protection against piping failures for System 80+ utilizes two criteria:

(1) justification, through application of leak-before-break (LBB), that the dynamic effects of specific piping system failures do not need to be considered.

(2) for pipe failures whose dynamic effects need to be considered, that plant design assures that these failures do not cause loss of safety-related systems needed to safely shut down the plant.

The "broad scope" rule of GDC-4 is implemented in applying LBB to meet the first criterion. Section 3.6.2 of CESSAR-DC (Amendment E) identifies the piping to which LBB is applied, and Section 3.6.3 describes the LBB methodology. The second criterion is met through the implementation of design features such as physical separation, barriers, pipe whip restraints and jet impingement shield, all of which are discussed in Section 3.6.1.3. of CESSAR-DC, Amendment E.

Question 410.50:

Identify the "potential hazards and highlighted susceptibilities" which are being developed, as stated, in SAR Section 3.6.1.2.C. Identify the design changes which have resulted from this ongoing review.

Response 410.50:

Reviews referred to in Section 3.6.1.2 (bottom of page 3.6-7) are part of the normal design process. Such reviews have been conducted for pre-System 80+ plants and have resulted in the implementation of design features such as redundancy, equipment layout, pipe whip restraints, jet impingement shields, and flood mitigation measures. These "lessons learned" have been applied to the System 80+ design where appropriate. At this time no known potential hazards or susceptibilities exist for the System 80+ design. The paragraph in CESSAR-DC will be clarified by deleting the reference to previous design efforts and will, instead, refer only to the review process for System 80+.

Question 410.51:

For the spent fuel storage cooling analysis of SAR Section 9.1.2.3:

(a) Discuss the spent fuel pool storage rack design features which enhance natural convection water circulation within the pool and adequate flow to all rack locations in the pool.

(b) Provide an evaluation of the thermal performance and hydraulic stability of the spent fuel storage racks for all postulated normal and accident conditions. Include analysis for a dropped fuel assembly which is reducing the flow area above fuel storage locations in the pool.

Response 410.51:

(a) The spent fuel pool storage racks have several design features to enhance natural circulation flow rate and flow distribution within the pool:

- Each fuel assembly cell wall in the rack module contains a coolant inlet hole at the bottom end of the cell (see Figure 9.1-1).

- Each fuel assembly cell in the rack module is open directly below the fuel assembly except for the perimeter cells which are closed off by a support plate.

- There is a vertical gap (approximately 1 inch high) between the floor surface of the spent fuel pool and the bottom end of the rack modules.

- The vertical gap, the bottom cell opening and the four wall inlet holes per cell provide multiple paths for the water to circulate from the perimeter of the spent fuel pool inwards to each fuel rack assembly cell.

(b) Analyses are performed for the spent fuel storage racks for normal and accident conditions in order to demonstrate that the thermal-hydraulic (T-H) criteria listed below are not violated:

- During normal operation bulk boiling will not exist in the pool

- Maximum fuel clad temperature will not exceed 650°F during normal operation or accident conditions.

The first T-H design criterion minimizes the potential for accelerated clad degradation associated with bulk boiling. This criterion also serves to minimize the release into the fuel storage building of fission gases that could leak into the spent fuel pool from failed fuel. The second T-H criterion assures that the spent fuel is not damaged by overheating. The design limit of 650°F is selected based on the fact that the fuel clad typically reaches this temperature in the reactor during normal operation; thus, fuel damage is unlikely if this temperature is not exceeded in the spent-fuel storage facility.

Thermal-hydraulic analyses are performed for the spent-fuel storage boxes to ensure that the T-H design criteria discussed above for the normal and accident conditions are met. To meet these criteria, normal operation is defined as a maximum pool bulk water temperature of 150°F at the fuel rack inlet and a minimum pool depth of 20 feet of water above the racks. Accident conditions are analyzed for two concurrent events, namely dropping a fuel assembly hoist box onto the stored fuel and a loss of cooling capability event. A loss of cooling event assumes that coolant is evaporated because of loss of external heat removal capability. The decay heat of the fuel is removed by boiling the pool water. The loss of cooling event is analyzed by assuming a pool depth of 10 feet of water above the racks and a maximum pool bulk temperature at the bottom of the racks equal to the saturation temperature at the top surface of the pool (212°F). The basis for using a 10 foot reduction in pool depth is that this value provides adequate time for taking action to restore the heat removal capability.

Thermal-hydraulic analyses are performed based on the process involving iteration on flow into the limiting fuel cell through a natural circulation flow resistance network until the natural circulation driving pressure is equal to the pressure losses through the network.

For the accident condition, it is assumed that, as a result of loss of external cooling, coolant is evaporated to a minimum pool depth of 10 feet of water above the storage racks. To meet the double contingency requirements, it is further assumed that the fuel cells are blocked by a dropped fuel hoist box. Based on a conservative estimate, this blockage extends over 90% of the total cell flow area. The additional pressure drop associated with this blockage retards flow into the fuel cell slightly. This is mainly due to the fact that most of the resistance to flow is provided by friction along the rods. Thus, the coolant temperature rise in a fuel cell is almost unaffected by this blockage.

Question 410.52:

Provide an evaluation of the containment design features which preclude any postulated leak or failure of the reactor cavity refueling pool seal or mitigate/preclude any level reduction in the spent fuel and refueling pools.

Response 410.52:

The reactor cavity refueling pool seal is permanently installed around the reactor vessel prior to initial fuel loading. The seal is designed to maintain its integrity during an SSE with the pool filled to the normal refueling water level. Therefore, fuel assembly cooling and radiation levels in the work area are maintained at acceptable levels.

During heavy load movement over the pool seal, e.g., reactor vessel head and reactor vessel internals, the fuel transfer tube valve is closed to insure against possible water level decrease in the fuel building in the event of a dropped load.

Question 410.53:

Section 9.1.3.3.2 discusses the possibility of an accidental opening of the gate between the spent fuel pool and a dry transfer canal and the resulting decrease in spent fuel pool level. Provide an evaluation on the effect of this reduced pool level on spent fuel pool pump operation in light of the elevated location of suction and discharge lines and NPSH requirements.

Response 410.53:

The design of the gate between the spent fuel pool and the fuel transfer system canal incorporates a hinge to allow the gate to be manually opened and closed. This feature eliminates reliance on an overhead crane for handling the gate and evaluating the consequences of dropping the gate on the spent fuel racks.

With a dry fuel transfer system canal, the resultant water pressure against the gate would prevent the manual opening of the gate (the gate opens into the spent fuel pool).

Since the accidental opening of this gate valve is not a credible accident, the related discussion was removed in Amendment I.

Question 410.54:

The safety evaluation of both the new and spent fuel storage areas includes an evaluation of the effects of dropping a fuel assembly and its handling tool from a height of two feet above the storage rack. Provide the following additional information in accordance with SRP 9.1.2, Item III.2.e guidance: Verify that the drop of any allowed lighter loads at a greater height does not result in a higher potential energy than a fuel assembly and its handling tool dropped from its normal operating elevation. Perform an evaluation of this in accordance with SRP 9.1.4 guidance.

Response 410.54:

The spent fuel racks have been evaluated and the results show that the rack k_{eff} will be less than .95 under the following postulated accident conditions:

(1) Drop of a fuel assembly handling tool from its maximum lift height over the fuel racks.

(2) Drop of a fuel assembly and the handling tool from their maximum lift height over the fuel racks.

(3) Drop of other items, such as a failed fuel canister with a fuel assembly, from their maximum lift height over the fuel racks.

Question 410.55:

Provide the following information in order to permit the staff to perform an integrated review of the spent fuel pool cooling and cleanup system (SFPCCS):

(a) Design parameters for major SFPCCS components (e.g., pumps, heat exchangers, tank, filters, demineralizers). Include the following minimum information on SFPCCS heat exchangers:

(1) Heat exchanger tube surface area (square feet),
(2) Heat exchanger conductance (Btu/ft^2-°F),
(3) Spent fuel pool water flow rate, per pump (Lb/Hr),
(4) Component cooling water flowrate, per heat exchanger (Lb/Hr), and
(5) Design component cooling water inlet temperature to the heat exchanger (°F),

(b) System interface requirements,

(c) SFPCCS design provisions which permit appropriate inservice inspection and functional testing as stated in SRP 9.1.3, Section III.1.g guidelines, and

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

(1) pool mixing,
(2) adequate system capacity,
(3) acceptable instrumentation and sampling capability,
(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:

(a) CESSAR-DC Section 9.1.3 describes the Pool Cooling and Purification System (PCPS) and provides the design bases for the system and for specific components. Compliance with these design bases during equipment procurement and construction will ensure that the safety function of the PCPS is accomplished. It is recognized that in previous Operating License reviews, detailed final design information was available for NRC staff review, however, for design certifications details for many subsystem components are not available until procurement is initiated.

(b) The System 80+ Standard Design encompasses an essentially complete plant. All systems connected to the PCPS and all associated structures are now within the scope of System 80+. The interfacing information has therefore been incorporated into the sections of CESSAR-DC which described those supporting systems.

(c) CESSAR-DC Section 6.6 discussed the in-service inspection of Class 2 and Class 3 components for components subject to examination.

CESSAR-DC Section 3.9.6 discusses the in-service testing program for Code Class 1, 2 and 3 pumps and valves. No special equipment tests are required since system components are normally in operation when spent fuel is stored in the pool.

(d) Features which enhance pool circulation and adequate flow to all rack locations in the pool are discussed in the response to Question 410.51.

The design bases for determining the system capacity is discussed in CESSAR-DC Section 9.1.3. Compliance with these design bases during equipment procurement will ensure that the safety function (pool cooling) is accomplished.

Question 410.56:

Provide the heat generation rate calculations using NUREG-0800, Standard Review Plan, Branch Technical Position ASB 9-2 and Section 9.1.3 guidance for the following cases:

(a) Normal refueling until the spent fuel pool is full, and
(b) same as case (a) above except the last available locations in the spent fuel pool are filled by a core offload.

Response 410.56:

The design bases for the heat loads are discussed in CESSAR-DC Section 9.1.3.1. The bases exceed (are more conservative than) the guidance of BTP ASB 9-2 and SRP Section 9.1.3.

The heat generation rate calculations for the spent fuel cooling design analysis are based on the ORIGEN 2 methodology. The residual decay heat release-vs-time curves generated with this methodology are comparable or conservative relative to NUREG-0800 Section 9.1.3.

The design analysis to show that the most limiting fuel assembly is adequately cooled without bulk boiling conservatively assumes that all allowed locations in the spent fuel racks are filled with fuel assemblies, and that each fuel assembly has the decay heat generation rate corresponding to the highest powered discharge fuel assembly following a full core offload, assuming 150 hours decay. This calculation is conservative relative to the requirements given in NUREG-0800, Section 9.1.3.

Question 410.57:

Explain the apparent discrepancy in the quantity of spent fuel stored as stated in SAR Section 9.1.2 and that in SAR Section 9.1.3.1.4 in accordance with SRP 9.1.3, Section III.1.c guidance. Also, explain how the design spent fuel storage capacity relates to the design life of the power plant and the expected number of fuel assemblies that are expected to be discharged/shipped to repository during this lifetime.

Response 410.57:

The configuration of the spent fuel racks that are provided to meet the minimum storage requirements of the SRP (1044 fuel assemblies) provides 1075 usable cavities. Consideration of the additional 31 fuel assemblies do not affect the calculations used to determine fuel pool water temperatures.

Since the plant design life is 60 years and the spent fuel racks can only accommodate 10 years of refueling discharges, approximately 4000 spent fuel assemblies will either have to be consolidated, stored in on-site dry casks, or shipped to a repository.

Question 410.58:

Provide the design information necessary to ensure that in the event failure of drains, inlets, outlets, or piping will not result in the spent fuel pool level inadvertently dropping below a point approximately ten feet above the top of the active fuel in accordance with SRP 9.1.3, Section III.1.e guidance.

Response 410.58:

The spent fuel pool cooling system has been designed to that the suction line is at least 10 feet above the top of the spent fuel racks. The discharge line, which penetrates the pool wall above the suction line and provides cooling water to spargers at the bottom of the pool, is equipped with a siphon breaker to preclude inadvertent pool draindown below the suction intake.

All other piping penetrations within the spent fuel pool are more than 10 feet above the top of the spent fuel racks.

Small floor drainlines, with double valve isolation, are provided in the spent fuel cask laydown area and the fuel transfer system canal to facilitate water removal from these areas after they have been isolated from the spent fuel pool by sealed gates.

Question 410.59:

Provide SFPCCS information which assures that leakage detection, component/header isolation capability, and inter-system leakage provisions are incorporated in this design per guidance of SRP 9.1.3, Section III.3.

Response 410.59:

CESSAR-DC Section 9.1.3 provides the description and design bases of the Pool Cooling and Purification System (PCPS). Please see the responses to question 410.555(a) for additional explanations.

Question 410.60 (9.1.2, 9.1.3):

Explain the discrepancy in stating that the maximum pool temperature is 150°F in SAR Section 9.1.2.3.5 and that the maximum pool temperature is 140°F in SAR Section 9.1.3.1.4.

Response 410.60:

Section 9.1.3.1.4 specifies that the maximum bulk water temperature (under heat load conditions of a full core offload with 10 years of irradiated fuel in the pool) is 140°F. The bulk water temperature applies to the aggregate volume of water in the spent fuel pool and is based on past experience which showed that 140°F is a practical limit to ion exchanger performance. This limit also serves to provide margin against the 150°F spent fuel rack design requirement.

The water temperature of 150°F specified in Section 9.1.2.3.5 describes the maximum design condition of the water at the fuel rack inlet flow passages, and is not indicative of the bulk water temperature in the spent fuel pool. Therefore, there is no discrepancy between the temperatures stated in Sections 9.1.3.1.4 and 9.1.2.3.5.

Question 410.61:

Provide an evaluation that assures that any failures in the nonsafety-related spent fuel pool cleanup and associated systems cannot affect the functional performance of any safety-related components or systems in accordance with SRP 9.1.3, Section III.5 guidance.

Response 410.61:

Amendment I provides a revised P&ID of the Pool Cooling and Purification System (Figure 9.1-3). Valving on this P&ID and on those for interfacing systems shows the isolation capability.

Question 410.62:

You have stated in SAR Section 9.1.3.2.1 that "The spent fuel pool receives normal borated water makeup from a water source .... In addition, the backup to the normal makeup system consists of piping and/or hoses from an alternate water source." Provide the detailed information concerning the normal makeup system and "alternate water source" makeup system including related technical data and cross-references. Also, update Figure 9.1-3, Spent Fuel Pool Cooling and Cleanup P&ID concerning the above makeup information.

Response 410.62:

The normal borated water source used to makeup to the spent fuel pool (to maintain water level within specified limits) is from the boric acid storage tank (BAST) in the Chemical and Volume Control System. The BAST meets the specified water chemistry requirements of the Pool Cooling and Purification System (PCPS) and is designed to Seismic Category I requirements. Refer to CESSAR-DC Amendment I, Figure 9.1-3, PCPS Piping and Instrumentation Diagram.

The backup to the normal source of makeup is the Station Service Water System. This system meets all requirements for an assured Seismic Category I backup water storage source. It is not permanently connected to the Pool Cooling and Purification System. The Station Service Water system is described in Section 9.2.1.

Questions 440.5:

In addition to assurance that k_{eff} is less than 0.98 with optimum moderation, the new fuel storage design bases should also include assurance that k_{eff} is less than 0.95 in the event the fuel area becomes fully flooded with full density unborated, pure water.

Response 440.5:

Detailed calculations for the new fuel storage design have been performed which confirm that k_{eff} is less than 0.95 in the event that the fuel area is fully flooded with full density unborated, pure water. The design analysis has the following results:

Condition	k_{eff}
Full load of unborated, full density water	0.9293
Optimum moderation	0.9458

The design calculations for k_{eff} are based upon a fuel enrichment of 5.0 wt% U-235 and include calculational uncertainties for the KENO-IV methodology.

Question 440.6:

The acceptability of the calculational methods (DOT-4 and KENO-IV) and the qualification of CE in their use should be documented either by including benchmark calculations performed by CE with these methods or by referencing previous NRC approval of CE use of these methods.

Response 440.6:

The benchmark analyses and methods uncertainties applied for criticality analyses for new fuel storage, spent fuel storage, and the refueling system are documented. NRC has previously approved the KENO-IV methodology used under Materials License SNM-1067. NRC has previously approved license amendments for spent fuel storage facilities which employed analyses based on the DOT-4 methodology (e.g. SER for Amendment 21 to Facility Operating License No. NPF-16, St. Lucie Plant, Unit No. 2).

Question 440.7:

Include a discussion of the method bias and uncertainty as well as other uncertainties considered such as those due to variations in the mechanical and material specifications from their nominal values. Verify that these uncertainties are combined will k_{eff} equivalent to a 95/95 probability confidence level for fuel storage calculations.

Response 440.7:

The uncertainty analyses for criticality calculations include components due to methodology and applicable uncertainties in dimension of structures, material tolerances, and temperature. The calculated uncertainty provides the equivalent of a one-sided 95/95 probability/confidence level in absolute reactivity units.

Question 440.8:

Explain why Paragraph C of Section 9.1.2.3.1.2 refers to an assumed boron concentration of at least 2000 ppm in the spent fuel pool in evaluating a dropped fuel assembly accident whereas Paragraph A of Section 9.1.2.3.1.3 implies that less than one-half of normal (about 1000 ppm) is assumed.

Response 440.8:

The expected boron concentration in the spent fuel pool of at least 2000 ppm imposes a lower bound value through the plant Technical Specification. The assumption of approximately half this value in criticality analyses for a dropped fuel assembly accident is used to provide a further measure of conservatism in analyzing criticality safety. Actual analyses have assumed 800 ppm and 1200 ppm for the most limiting dropped fuel assembly event, showing that k_{eff} remains below the acceptance level in these cases.

Question 440.9:

Explain what is meant in Section 9.1.2.3.1.3 by "borated" or "mixed" modes and which neutron absorption effects is credit taken for. This paragraph also seems to imply a two-region pool with burnup credit allowed. However, this is not described in the spent fuel pool storage rack description in Section 9.1.2.2.2.

Response 440.9:

In Section 9.1.2.3.1.3 "mixed modes" refers to use of two storage regions for spent fuel as described in Section 9.1.2.2.2. Neutron absorption effects are credited if a freshly burned fuel assembly which the plant Technical Specifications require to be stored in Region I is instead inadvertently placed in Region II. The "borated" mode refers to the provisional feature of the structural design of the spent fuel storage racks but would allow accepting a 100% storage arrangement with neutron poison inserts, as indicated in Section 9.1.2.2.2. This is a provisional structural feature only, however, and is not provided for in the analysis supporting this design or in the Technical Specifications.

Questions 471.1-4, 471.9, 471.11, 471.12, 471.13:

Responses 471.1-4, 471.9, 471.11, 471.12, 471.13:

These questions have been resolved by Amendment I. [Note that numbers 471.5, 471.6, 471.7, and 471.10 were not used in this series of questions.]

Question 471.8:

In Table 12.2-1, Maximum Neutron Spectra Outside Reactor Vessel, Column: Average Neutron Energy (Mev), it appears that the neutron energy, 3.3 x 10, is incorrect - please verify.

Response 471.8:

The correct value is 3.3×10^{-1}. Table 12.2-1 will be revised in a future amendment.

Question 471.14:

Section 12.2.1.1.5, Chemical and Volume Control System (CVCS), Paragraph B.1 states, in part, that "All nuclides except Xe, Kr, Rb, and Cs have a decontamination factor (DF) of 10 and efficiency of 90%, Xe and Cs have (DF) of 1.0 and efficiency of 0%, Rb and Cs have a DF of 2.0, and efficiency of 50%."

Please review the accuracy of DF and efficiency for Cs, and specify DF and efficiency for Kr.

Response 471.14:

In Amendment I to CESSAR-DC, the error has been corrected (Cs changed to Kr) and the quoted sentence has been changed to read "All nuclides except Xe, Kr, Rb and Cs have a decontamination factor (DF) of 1[illegible] an efficiency of 90%, Xe and Kr have a DF of 1.0 and effici[illegible] 0%, Rb and Cs have a decontamination factor of 2.0 and an eff[illegible]f 50%.

Question 471.15:

Section 12.2.1.1.5, CVCS, please justify the large difference between the data quoted in B.1 and B.2 for Rb and Cs.

Response 471.15:

One of the factors which affects the decontamination factor (DF) and efficiency of an ion exchanger is the concentration in the incoming stream. The preholdup ion exchanger is downstream of the purification ion exchangers. The resulting concentration of Rb and Cs entering the preholdup ion exchanger results in the DF and efficiency given in B.2.

Question 500.13:

The Commission's Severe Accident Policy Statement included the policy that:

> "The issues of both insider and outsider sabotage threats ... will be emphasized in the design and in the operating procedures developed for new plants." (Emphasis added.)

Also, NUREG/CR-2643, "A Review of Selected Methods for Protecting Against Sabotage by an Insider," concluded that effective insider protection will require an integrated approach that includes the best features of (1) physical protection measures, (2) damage control measures, and (3) plant design measures. The physical protection measures studied in that report all involved some impact on site work rules and procedures.

However, CESSAR-DC Revision E Section 13.5, Plant Procedures, states that "the site operator's plant procedures is within the site operator's scope and shall be provided in the site-specific SAR." Such a blanket statement seems to remove the possibility of including procedural constraints as a part of the standard design sabotage protection design philosophy.

Response 500.13:

The referenced statement does not preclude any procedural constraint which might be imposed as part of the System 80+ design. Any procedural guidance would be stated in CESSAR-DC and provided to the utility in procedure guidelines. This guidance would, therefore, be an input to the detailed site procedures developed by the owner operator.

Question 500.14:

Section 1.2.13, Physical Plant Security and Protection From Sabotage," states that these design features are described in Chapters 2, 3, 7, 8, and 9. To assist in our review, please specify where in those chapters we should find such features.

Response 500.14:

Section 1.2.13, Amendment E, now references Appendix A to Chapter 13 for a discussion of design features which inhibit sabotage. The statement referenced in the question has been removed from CESSAR-DC.

PROPOSED REVISIONS TO THE COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT - DESIGN CERTIFICATION

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a. Penetration assemblies

b. Isolation valves

c. Equipment hatch

d. Emergency personnel hatch

e. Personnel lock

f. Liner plate

g. Test connections

h. Piping between penetration assemblies and isolation valves.

17. Diesel Generator Building HVAC System.

18. Emergency Feedwater System.

19. Condensate Storage System.

20. Ex-core Neutron Monitoring.

21. Station Service Water System.

22. Air Coolers.

E

C. For other postulated breaks not included in items A and B above, systems must not be affected such that any break, evaluated on a case-by-case basis, violates the following criteria:

1. The pipe break must not cause a reactor coolant, steam, or feedwater line break.

2. The function of safety systems required to perform protective actions to mitigate the consequences of the postulated break must be maintained.

3. The ability to place the plant in a safe shutdown condition must be maintained.

A systematic integrated review of ~~[illegible]~~ safety-related and associated systems ~~[illegible]~~ is conducted during the design process to verify compliance with design criteria, interface requirements, and safety design bases. ~~[illegible]~~

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~~equipment from common mode failures, as well as provided an independent method of verification of the availability of essential equipment required to mitigate the consequences of postulated accident scenarios. The resolution of comments raised during these reviews resulted in changes to equipment layout, design of pipe whip and jet impingement restraints, upgrading some non-seismic supports to seismic, and the addition of curbs, drains, and other flood mitigation measures.~~

The potential effects of flooding as a consequence of a pipe break, or leakage or through-wall cracks (as defined in Sections 3.6.2.1.2.C and 3.6.2.1.2.D) were analyzed on a case-by-case basis to ensure that the operability of safety-related equipment would not be impaired.

An analysis of the potential effects of missiles is discussed in Section 3.5.

The potential environmental effects of steam on essential systems are discussed in Section 3.11. In general, because of the protective measures of redundancy and separation between systems and trains, the consequential effect of the transport of steam will not be sufficient to impair the ability of the essential system to shut down the plant and/or mitigate the consequences of the given accident of interest. E

There are no high-energy lines in the vicinity of the control room. As such, there are no effects upon the habitability of the control room by pipe break either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6.1.3 Safety Evaluation

By means of design features such as separation, barriers, and pipe whip and jet impingement restraints, all of which are discussed below, the effects of pipe break will not damage essential systems to an extent that would impair their design function nor affect necessary component operability.

The ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event is discussed in the failure modes and effects analyses provided in Sections 5.4.7, 6.2, 6.3, 6.5, 7.2, 7.3, 8.3, 9.2 and 10.4.

A. Separation

The plant arrangement provides separation to the extent practical between redundant safety systems in order to

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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.

The design flow rate and filtering capability of the SFPCCS 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).

The design flow rate of the SFPCCS shall provide at least two complete water changes per day for the entire volume of the spent fuel pool.

The SFPCCS shall maintain the refueling pool, spent fuel pool, and IRWST (PWR) water chemistry and clarity within the limits specified below:

- Conductivity less than [illegible] μsiemens/cm @ 25°C;
- pH between 4.5 and 10 @ 25°C,
- Chlorides less than 0.15 ppm, and
- Optical clarity less than 1.0 [illegible] turbidity.

Insert

9.1.3.4 Tests and Inspections

Components of the spent fuel pool cooling and cleanup system are in either continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. The Seismic Category I portions will be inspected in accordance with the ASME B&PV Code, Section XI.

No special equipment tests are required since system components are normally in operation when spent fuel is stored in the fuel pool

Sampling of the fuel pool water is performed for gross activity and particulate matter concentration. The layout of the components of the SFPCCs is such that periodic testing and inservice inspection of this system are possible.

9.1.3.5 Instrumentation Application

The instrumentation provided for the spent fuel pool cooling and cleanup system is discussed in the following paragraphs. Alarms and indications are provided as noted.

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Spent fuel pool and demineralizer effluent will be monitored by grab samples with laboratory analysis. The fuel pool will be monitored to ensure that the water quality is maintained within the above limits.

Demineralizer replacement is to be based on three criteria:

1. Breakthrough of cesium, cobalt, chloride, or fluoride.

2. Pressure drop not to exceed demineralizer and resin vendors' recommended limits for the as-procured equipment.

3. Thermal expansion approaching resin vendors' recommended limit for the as-procured equipment.

TABLE 12.2-1

MAXIMUM NEUTRON SPECTRA OUTSIDE REACTOR VESSEL[(a)]

Average Neutron Energy (Mev)	Neutron Spectra (neutrons/cm^2-s)
13.60	$5.90 \times 10^{+6}$
11.10	$1.86 \times 10^{+7}$
9.10	$3.79 \times 10^{+7}$
7.27	$6.87 \times 10^{+7}$
5.66	$1.08 \times 10^{+8}$
4.51	$9.08 \times 10^{+7}$
3.53	$1.58 \times 10^{+8}$
2.73	$2.01 \times 10^{+8}$
2.40	$6.69 \times 10^{+7}$
2.09	$3.86 \times 10^{+8}$
1.47	$1.48 \times 10^{+9}$
8.30×10^{-1}	$5.20 \times 10^{+9}$
3.30×10^{-1}	$1.47 \times 10^{+10}$
5.70×10^{-2}	$1.05 \times 10^{+10}$
1.96×10^{-3}	$3.24 \times 10^{+9}$
3.42×10^{-4}	$2.96 \times 10^{+9}$
6.50×10^{-5}	$2.01 \times 10^{+9}$
1.98×10^{-5}	$1.27 \times 10^{+9}$
6.90×10^{-6}	$1.44 \times 10^{+9}$
2.09×10^{-6}	$1.09 \times 10^{+9}$
7.60×10^{-7}	$9.67 \times 10^{+8}$
2.50×10^{-8} (thermal)	$6.22 \times 10^{+9}$

(a) At core midplane, one half foot from vessel surface

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13.3.3.3 Operations Support Center

13.3.3.3.1 Summary Description

The Operations Support Center (OSC) is an onsite facility separate from the Control Room and the Technical Support Center where operations support personnel will assemble in an emergency. The OSC is located in the Control Complex, above the Control Room. This facility is not specifically required by 10 CFR 50 Appendix A, but is included in this plant design to ensure that an adequate facility is provided for onsite emergency maintenance and other personnel to gather as a ready resource to support actions initiated by the Control Room. There is a direct communications link between the Control Room and the OSC so that all personnel reporting to the OSC can be assigned to duties in support of emergency operations.

Until such time as the Operations Support Center is activated, all functions of this facility are performed in the Control Room. The OSC is activated based on the emergency class and the specific conditions surrounding an accident. The activation and use of the OSC is specified in the emergency planning section of the site-specific SAR. OSC staffing levels will depend on the severity of the emergency condition; these are also addressed in the emergency planning section of the site-specific SAR.

13.3.3.3.2 Function

The Operations Support Center provides two main functions. The OSC:

- Provides a location where plant logistic support can be coordinated during an emergency.
- Restricts control room access to those support personnel specifically requested by the shift supervisor.

13.3.3.3.3 Location

The Operations Support Center is located in the control complex, above the control room.

13.3.3.3.4 Habitability

Operations Support Center personnel are protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions, by the Nuclear Annex general building ventilation system. Therefore, the OSC protection level is the same as that of the rest of the building, with the exception of the Technical Support Center and the Control Room

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which are covered by a different system. Should the OSC become uninhabitable, the OSC functions can be performed by essential support personnel in the control room or other designated on-site locations. Reference the site-specific SAR for emergency plans and habitability details.

13.3.3.3.5 Communications

Since the function of the OSC is one of support to emergency operations, the OSC has direct communications with the control room and the TSC. This ensures that personnel reporting to the OSC can be assigned to duties in support of emergency operations.

The OSC communications system consists of:

- One dedicated telephone line to the control room
- One dedicated telephone lint to the TSC
- Dial telephones that provide access to onsite and offsite locations

The OSC communications system may also include direct voice and/or radio intercommunications links as backup or supplementary communications means. Refer to the site-specific SAR for detailed information on the applicability and use of these other means of communications.

13.3.3.4 <u>Laboratory Facilities</u>

13.3.3.4.1 Summary Description

Consistent with the guidance stated in NUREG-0654, II.H.9 and NUREG-0737, II.B.3, the System 80+ Standard Plant design makes provisions for both HOT and conventional Laboratory Facilities. The hot facilities are currently shown to be located in the Radwaste Building and the Nuclear Annex. Space for a large conventional laboratory is provided in the water chemistry building. Locations for other, smaller lab facilities are provided at various places throughout the plant. The laboratories are provided to support efforts to monitor plant systems and environmental samples for compliance with technical specifications.

13.3.3.4.2 Function

The primary functions of the laboratories are:

- to provide plant support services for routine analyses required for personnel protection, surveys, and related health physics functions

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- to provide normal and post-accident cold chemical analyses on required plant chemistry samples
- to provide routine and post-accident counting on all plant radioactivity samples
- to provide grab sample analyses used as a check on the accuracy of the continuous on-line process monitoring instrumentation
- to provide a facility to store and secure radioactive calibration and check sources and instruments undergoing calibration, maintenance, or repair

13.3.3.4.3 Location

The hot laboratory facilities are currently shown to be located in the Radwaste Building and the Nuclear Annex. Space for a large conventional laboratory is provided in the water chemistry building. Radiation counting rooms and instrument calibration areas are located at elevation 115+6 in the Nuclear Annex Outage/Maintenance Area. Locations for other, smaller lab facilities are provided at various places throughout the plant. Locations for these facilities are provided to assure that all critical onsite sampling capabilities (see Regulatory Guide 1.97) can be performed to the required accuracy at the plant site, and such that ALWR normal and post-accident sampling requirements are met.

13.3.3.4.4 Features

In order to meet the intent of the aforementioned guidance, the CESSAR-DC laboratory facilities are designed with the following features:

- adequate space for expansion to accommodate changes in available technology and equipment
- radiation counting rooms, instrument calibration areas and checkout areas are located in low radiation zones and provided with shielding to reduce background radiation "noise"
- secured access to radioactive calibration and check sources

Sampling methods and instrumentation are discussed in the site-specific plant operations manuals. General maintenance is described in other plant operating documents.

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13.3.3.5 Post Accident Sampling

Consistent with the guidance stated in NUREG-0737, II.B.3, the System 80+ Standard Plant design provides for a Post Accident Sampling System. This system is located in the CVCS panel. System functions and design requirements are covered in Section 9.3.2.

13.3.3.6 Onsite Decontamination Facilities

13.3.3.6.1 Summary Description

The Onsite Decontamination Facilities (ODF) are onsite facilities located in the Nuclear Annex (el. 91+9) and in the Radwaste Facility. These facilities are provided to remove or reduce radioactive contaminants from plant equipment, protective clothing, and personnel. These facilities are to be designed according to particular client preference, but are to be supplied by the major decontamination equipment, including various spray nozzle assemblies, chemical and/or abrasive supply systems, collection and storage tanks, high pressure pumps, filters, demineralizers and piping connections to waste processors.

Included in the ODF are the hot laundry facilities, hose washdown stations, personnel decontamination fixtures, hot shower, radiation detection equipment and personnel decontamination supplies. Also included is equipment necessary to decontaminate small tools and instruments as well as larger tools and pieces of equipment.

These facilities are designed to meet the requirements as stated in 10 CFR 50 Appendix E, IV.E.3 and 10 CFR 50.47 (b) (8). The role(s) of the Onsite Decontamination Facilities in the event of a plant emergency shall be contained in the emergency planning section of the site-specific SAR.

13.3.3.6.2 Function

The functions of the Onsite Decontamination Facilities are:

- To facilitate equipment disposal by reducing contamination and radiation levels to releasable limits.
- To facilitate equipment repair by reducing contamination and radiation levels consistent with ALARA.
- To provide a location and supplies for personnel decontamination.

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13.3.3.6.3 Location

Onsite Decontamination Facilities are located as follows:

- Personnel Decontamination Facilities - Personnel decontamination areas are located in the Nuclear Annex. There are facilities at both the upper and lower personnel access portals to the containment.

- Equipment Decontamination Facilities - Equipment decontamination facilities are located in the Nuclear Annex (el. 91+9) and the Radwaste Facility (RWF). The hot laundry facilities are located in the RWF.

13.3.3.6.4 Features

The CESSAR-DC Onsite Decontamination Facilities are provided in full compliance with 10 CFR 50, Appendix E, IV.E.3 and 10 CFR 50.47 and ALARA considerations. As such, the following are included in the design of the facilities:

- Sinks, workbenches, and decontamination supply cabinets

- Alarmed radiation monitors near tanks, filters, demineralizers, etc. which are used in the decontamination processes

- Clean, adequate areas and provisions for staging, decontamination and checkout for applying and removing protective materials

13.3.3.6.5 Decontamination Methods and Procedures

Selection of decontamination methods to be employed in the Onsite Decontamination Facilities at a specific generating plant is the responsibility of the individual licensee. Some of the decontamination requirements may be met by using portable or otherwise transportable facilities at the discretion of the individual licensee.

Decontamination and radwaste control procedures are considered to be a fundamental part of the plant operations documentation. The individual, site-specific plant operations documents will contain these detailed procedures.