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ASEA BROWN BOVERI

February 14, 1992
LD-92-020

Docket No. 52-002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 205ʹ5

Subject: Response to NRC Requests for Additional Information

References: A) Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated February 15, 1991

B) Letter, Reactor Systems Branch RAIs, T.V. Wambach (NRC) to E. H. Kennedy (C-E), dated May 13, 1991

C) Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated August 21, 1991

Dear Sirs:

The above References 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 References 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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Enclosure 1 to
LD-92-020

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION
REACTOR SYSTEMS BRANCH

QUESTION 440.41

CESSAR-DC Section 5.2.2.4.3.3 states that the main steam safety valves (MSSVs) are designed to operate in the environmental conditions with the maximum temperature of 330°F for 3 minutes following a main steam line break accident. Provide a temperature profile for the compartment housing the MSSVs during a design basis main steam line break accident to support the assumptions made in the environmental conditions.

RESPONSE 440.41

Each steam generator has its own main steam valve compartment housing. If a main steam line breaks inside one of the main steam valve compartments, the pressure of the associated steam generator would drop. Consequently, the MSSVs on the affected steam generator would not be called upon to open. After the affected steam generator blows down, decay heat would be removed via MSSVs and safety related ADVs on the unaffected steam generator. Since the affected main steam valve compartment does not interact with the intact main steam valve compartment, the temperature in the intact compartment will always remain below 330°F.

Question 440.42

The statement made in CESSAR-DC section 5.2.2.10.1.1 regarding operator action for low temperature overpressure protection (LTOP) is not clear. Discuss operator actions necessary during transients involving LTOP, including instrumentation and operating procedures available that ensure proper operator actions for mitigation of the transients.

Response

The operator action referred to in CESSAR-DC section 5.2.2.10.1.1 concerns an assumption that is made in the mass and energy addition transient analysis discussed in section 5.2.2.10.2.1. The assumption is strictly made for analysis and does not suggest any requirements by the operator or limitations on equipment.

Since the results of these transient analyses show that the system pressure reaches an equilibrium within several minutes, the assumption of no operator action for 10 minutes in the analysis is reasonable.

Question 440.44:

Provide the results of the analyses for the design basis mass addition and energy addition transients including transient curves that demonstrate the peak RCS pressures are within pressure temperature limits determined for the System 80+ design. Instrumentation uncertainties should be factored into your evaluation.

Response 440.44:

The results of the design basis mass addition and energy addition transients are attached and will be included in CESSAR-DC Chapter 5 section 5.2.2.10. These figures will appear in a future amendment to CESSAR-DC.

RAI 440.44

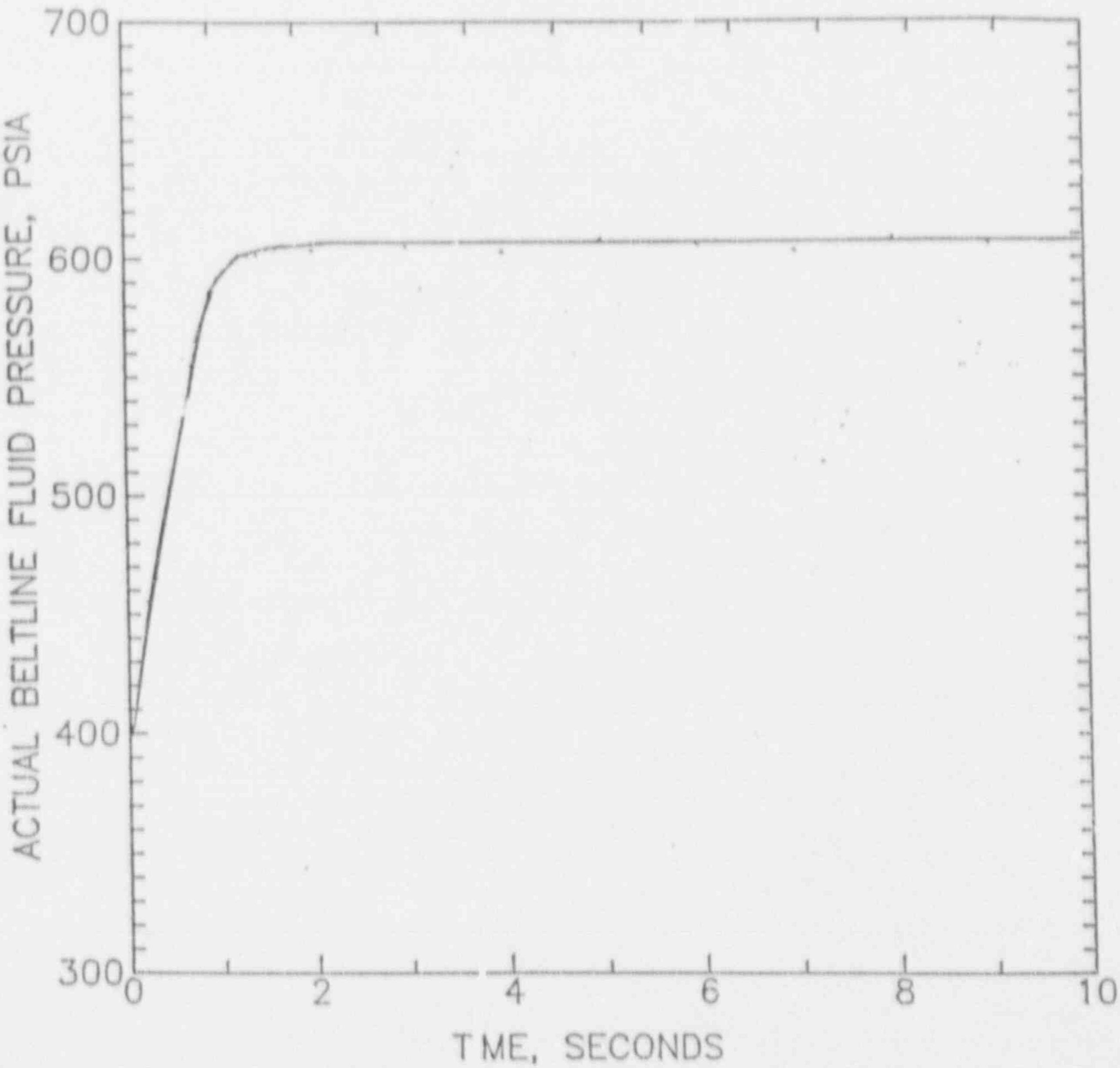

Figure 5.2-1 SYSTEM 80+ MASS ADDITION TRANSIENT (INADVERTENT SAFETY INJECTION ACTUATION)

RAI 440.44

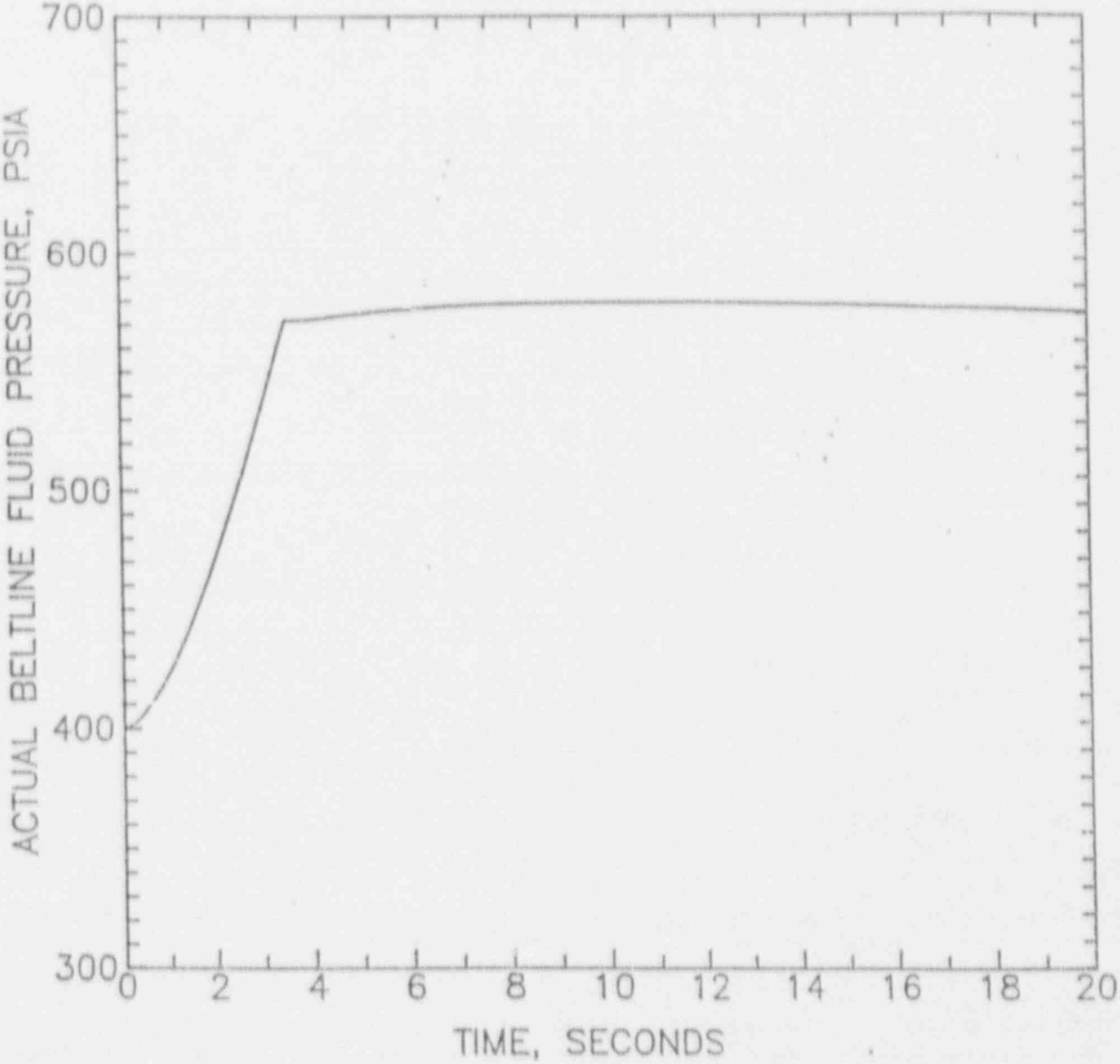

Figure 5.2-2 SYSTEM 80+ ENERGY ADDITION TRANSIENT (RCP START WITH RCS ΔT)

5.2.2.10.2.1 Limiting Transients

Transients during the low temperature operating mode are more severe when the RCS is operated in the water-solid condition. Addition of mass or energy to an isolated water-solid system produces increased system pressure. The severity of the pressure transients depends upon the rate and total quantity of mass or energy addition. The choice of the limiting LTOP transients is based on evaluations of potential transients for System 80 plants and their applicability to the System 80+ plant. The most limiting transients initiated by a single operator error or equipment failure are:

A. An inadvertent safety injection actuation (mass addition).

B. A reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy addition).

The most limiting transients are determined by conservative analyses which maximize mass and energy additions to the RCS. In addition, the RCS is assumed to be in a water-solid condition at the time of the transient; such a condition has been noticed to exist infrequently during plant operation since the operator is instructed to avoid water-solid conditions whenever possible.

Figure 5.2-1

~~Table 5.2-1~~ shows the results of the inadvertent safety injection actuation transient analysis for a water-solid RCS, when the RCS in the LTOP mode. The mass addition due to the simultaneous operation of four safety injection pumps and one charging pump was considered, along with the simultaneous addition of energy from decay heat and the pressurizer heaters.

Figure 5.2-2

~~Figure 5.2-1~~ shows the result of the transient analysis of reactor coolant pump start when a steam generator to reactor vessel ΔT of 100°F exists. This ΔT is the maximum allowed by technical specification during the LTOP mode. In addition to considering the energy addition to the RCS from the steam generator secondary side, energy addition from decay heat, the reactor coolant pump and all pressurizer heaters were also included. In this analysis the steam generators were assumed to be filled to the zero power, normal water level. For conservatism, the secondary water, both around and above the U-tubes, was assumed to be thermally mixed in order to maximize the energy input to the primary side. This assumption is conservative since as a result of the temperature distribution within the steam generator during the transient, the water inventory above the tubes is practically isolated thermally from the heat transfer region. Therefore the heat transfer rate, and thus the primary side pressure, is not sensitive to the secondary side water level as long as the tubes are covered.

On the basis of experience, the ΔT value of 100°F used in the analysis is much larger than any ΔT that might be expected during plant operation. This maximum allowable ΔT of 100°F will prevent pressurizer pressure from exceeding the minimum P-T limit allowed for the lowest system temperature during the LTOP mode of operation. (See Figures 5.3-5a and 5.3-5b). During RCS cooldown | E using the Shutdown Cooling System, coolant circulating with the reactor coolant pumps operating serves to cool the steam generator to keep the temperature difference between the reactor vessel and the steam generator minimal. Procedures for System 80+ have directed the operator to maintain the ΔT below approximately 20°F.

LTOP transients have not been analyzed for the simultaneous startup of more than one reactor coolant pump (RCP). Such operation is procedurally precluded since the operator starts only one RCP at a time and a second RCP is not started until system pressure is stabilized. Additionally, there is an LTOP transient alarm that should indicate that a pressure transient is occurring. Accordingly, the second RCP would not be started.

The operator cannot start an RCP if the ΔT exceeds 100°F. However, as mentioned above, administrative procedures for System 80 have ensured that the ΔT is maintained below approximately 20°F. With similar administrative controls on System 80+, ΔT margins will be even greater that for System 80. E

Figures

The results of the analyses provided in Figure 5.2-1 and Table 5.2-2 5.2-1 show that the use of either SCS relief valve will provide sufficient pressure relief capacity to mitigate the most limiting LTOP events identified above.

5.2.2.10.2.2 Provision for Overpressure Protection

During heatup, the RCS pressure is maintained below the LTOP pressure until the RCS cold-leg temperature exceeds the LTOP disable temperature. During cooldown, the RCS pressure is maintained below the LTOP pressure once the RCS cold-leg temperature reaches the LTOP enable temperature.

An LTOP enable temperature is defined in Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," to Standard Review Plan Section 5.2.2, "Overpressure Protection," issued November 1988 as Revision 2. The definition is based on measuring the degree of protection provided by the low temperature overpressure protection system (LTOP System) against violations of the P-T Limits in terms of the RT_{NDT} of the reactor vessel beltline material at either the 1/4t or 3/4t location, E

During cooldown, whenever the RCS cold leg temperature is below the LTOP enable temperature, that corresponding to the intersection of the design P-T Limit curve for cooldown with pressurizer safety valve setpoint, the SCS relief valves provide the necessary overpressure protection. If the SCS is not aligned to the RCS before the cold-leg temperature is decreased below the LTOP enable temperature, an alarm will notify the operator to open the SCS suction isolation valves. However, the SCS cannot be aligned to the RCS until the RCS pressure is below the LTOP enable pressure. E I E

The LTOP conditions described above are within the SCS operating range. Technical Specification Section 16.3/4.4.8.3 requires the SCS suction line isolation valves to be open when operating in the LTOP mode. Also, this Technical Specification ensures that appropriate action is taken if one or more SCS relief valves are out of service during the LTOP mode of operation.

Either SCS relief valve will provide sufficient relief capacity to prevent any pressure transient from exceeding the isolation interlock setpoint (see Figure 5.2-1 and ~~Table 5.2-3~~ Figure 5.2-2). E

5.2.2.10.2.3 Equipment Parameters

The SCS relief valves are spring-loaded liquid relief valves with sufficient capacity to mitigate the most limiting over-pressurization event. Pertinent valve parameters are as follows:

Parameter

Nominal Setpoint ~~550~~ 545 psia*
Accumulation 10%
Capacity ~~4000~~ 5000 (@10% acc) gpm E

Since each SCS relief valve is a self actuating spring-loaded liquid relief valve, control circuitry is not required. The valve will open when RCS pressure exceeds its setpoint.

The SCS relief valves are sized, based on an inadvertent safety injection actuation signal (SIAS) with full pressurizer heaters operating from a water-solid condition. The analysis assumes simultaneous operation of four SIS pumps and one charging pump with letdown isolated. The resulting flow capacity requirement for water is ~~3934~~ 5000 gpm. The analysis in Section 5.2.2.10.2.1 assumed that either SCS relief valve relieved water at this rate. I E

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* Pressure measured at the valve inlet. E

The design relief capacity of each of two SCS relief valves (shown in P&ID Figure 6.3.2-1B) as supplied by the valve manufacturer meets the minimum required relief capacity of (5000) 4000 gpm which contains sufficient margin in relieving capacity for even the worst transient. The SCS relief valves are Safety Class 2, designed to Section III of the ASME Code. E

5.2.2.10.2.4 Administrative Controls

Administrative controls necessary to implement the LTOP provisions are limited to those controls necessary to open the SCS isolation valves.

During cooldown, when the temperature of the RCS is above that corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves, and no administrative procedural controls are necessary. Before entering the low temperature region for which LTOP is necessary, RCS pressure is decreased to below the maximum pressure required for LTOP. The LTOP pressure is less than the maximum pressure allowable for SCS operation. Once the SCS is aligned, no further specific administrative procedural controls are needed to ensure proper overpressure protection. The SCS will remain aligned whenever the RCS is at low temperatures and the reactor vessel head is secured or until an adequate vent has been established. As designated in Table 7.5-2, indication of SCS isolation valve position is provided. E I

During heatup, the SCS isolation valves remain open at least until the LTOP enable temperature. Once the RCS temperature has reached that temperature corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves. The SCS can be isolated and no further administrative procedural controls are necessary. E

5.2.2.11 Pressurized Thermal Shock

The System 80+ reactor vessel meets the requirements of 10 CFR 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events." The calculated RT_{PTS} is 109°F which satisfies the screening criteria in 10 CFR 50.61(b)(2).

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specification

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are part of the reactor coolant pressure boundary is given in Table 5.2-2.

TABLE 5.2-4

RESULTS OF THE INADVERTENT SAFETY INJECTION ACTUATION TRANSIENT ANALYSIS[a] (FOR A WATER-SOLID RCS)

Number of Charging Pumps	Number of SIS Pumps	Number of SCS Relief Valves Operational	Maximum RCS Pressure (psia)	Maximum Relief Flowrate (gpm)
1	1	1	567	1260
1	2	1	580	2162
1	3	1	592	3055
1	4	1	608	3934

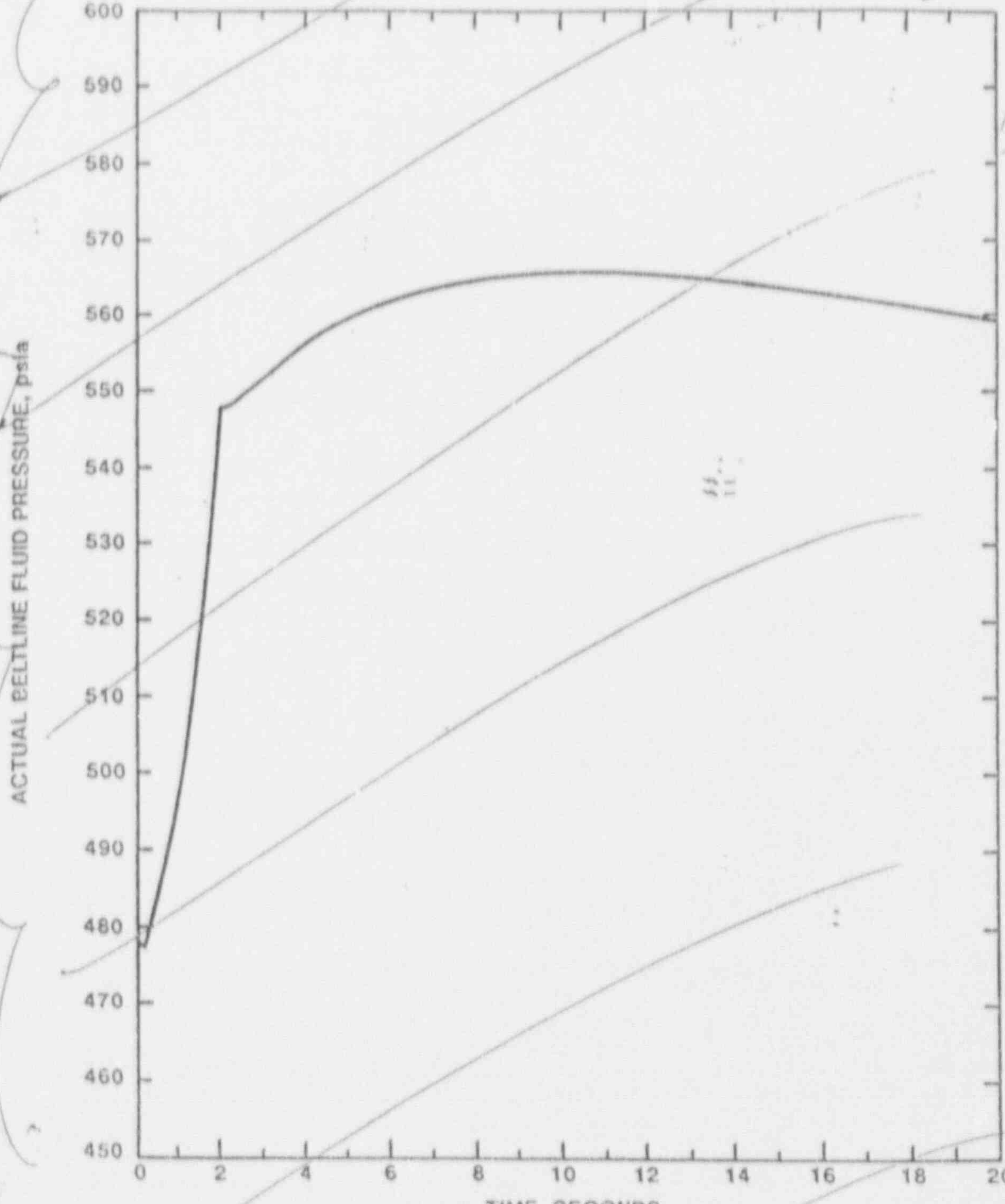
E

NOTE: a. Including the considerations of energy addition from decay heat and all pressurizer heaters.

Replace with new Figure 5.2-1

Replace with new Figure 5.2-2

RAI 446.44



Amendment E
December 30, 1988

SYSTEM 80+™

SYSTEM 80+ RCP START TRANSIENT

Figure

RAI 440.44

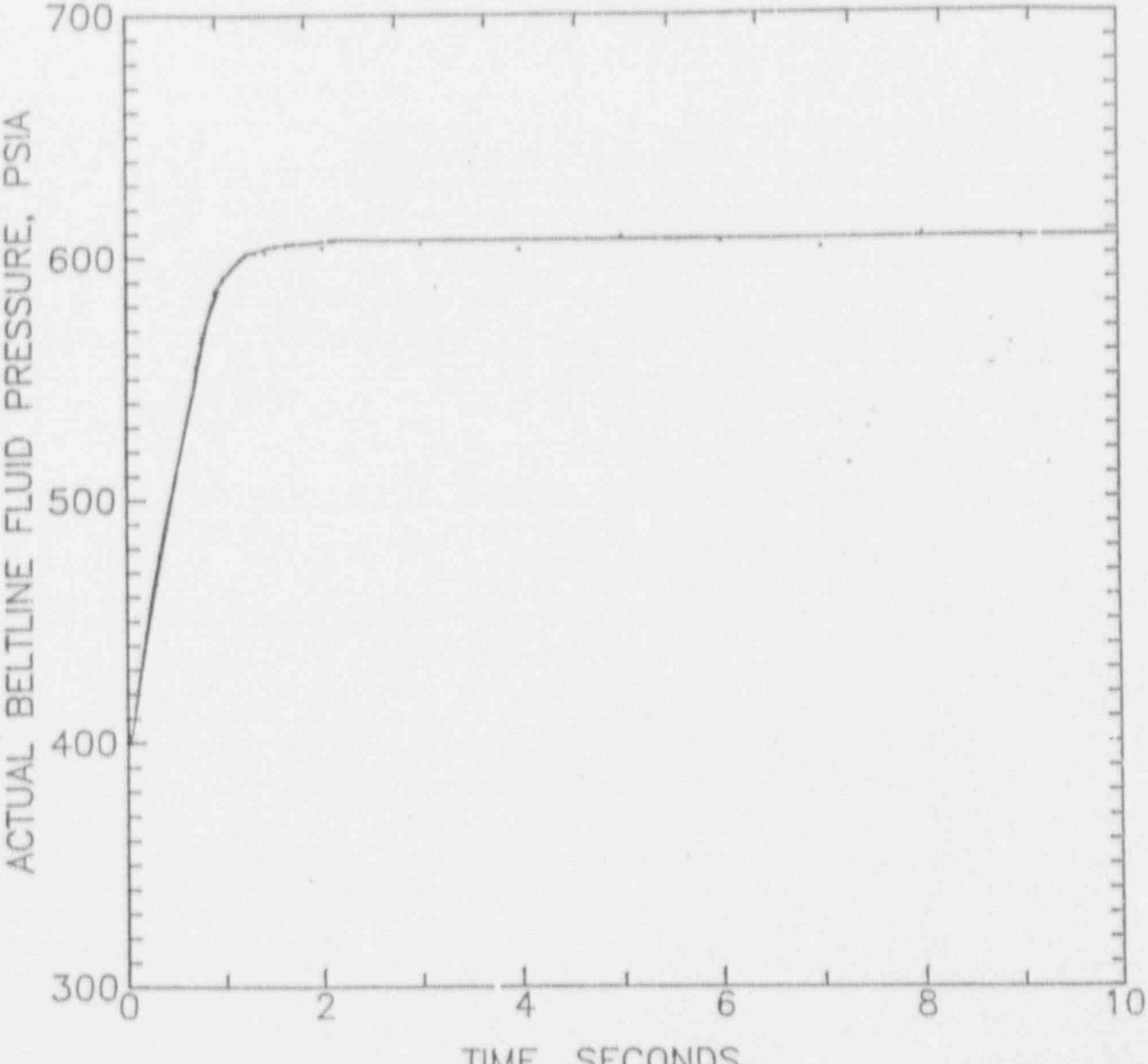


Figure 5.2-1 SYSTEM 80+ MASS ADDITION TRANSIENT (INADVERTENT SAFETY INJECTION ACTUATION)

RAI 440.44

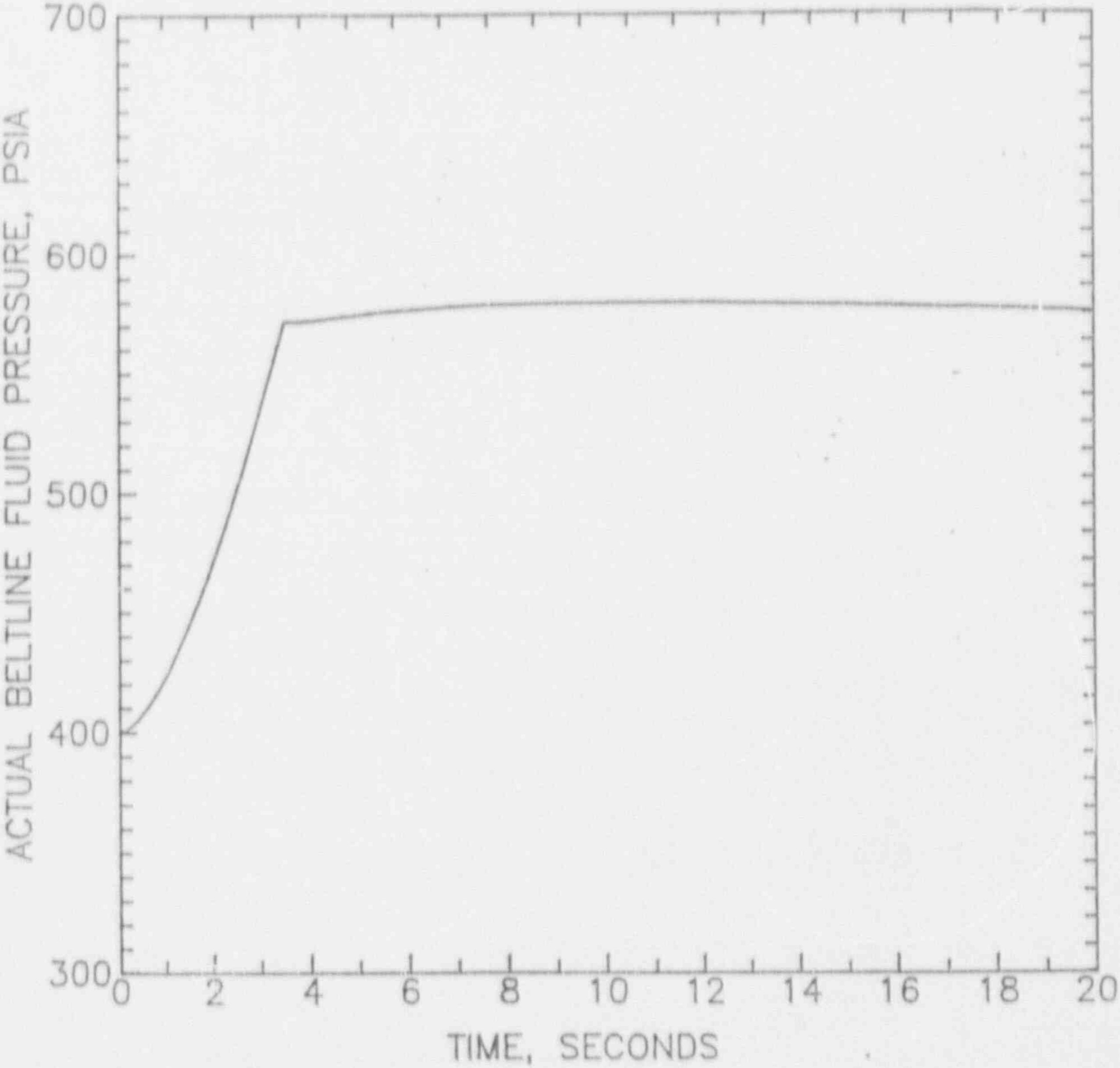


Figure 5.2-2 SYSTEM 80^{+} ENERGY ADDITION TRANSIENT (RCP START WITH RCS ΔT)

Question 440.45

The following RAI clarifies the staff's position regarding intersystem LOCA protection and supersedes RAI 440.17 which should be deleted.

Future evolutionary ALWR designs should reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing to the extent practicable all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength at least equal to full RCS pressure.

The "extent practicable" phrase is a realization that all systems must eventually interface with atmosphere pressure and that for certain large tanks and heat exchangers it would be difficult or prohibitively expensive to design such systems to an ultimate rupture strength equal to full reactor system pressure.

It should be noted that the degree of isolation or number of barriers (for example three isolation valves) is not sufficient justification for using low pressure components that can be practically designed to the ultimate rupture strength criteria. For example, piping runs should always be designed to meet the ultimate rupture strength criteria, as should all associated flanges, connectors, packings including valve stem seals, pump seals, heat exchanger tubes, valve bonnets and RCS drain and vent lines. The designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

Our initial review of System 80+ design features, including proposed resolution of generic safety issue GI 105, does not provide adequate information on how these systems will satisfy the above staff position for evolutionary ALWRs. Please provide detailed discussion of how the System 80+ design meets the above criteria. As part of the response include:

(1) an identification of all interfaces to the RCS indicating design and ultimate pressure capabilities for these systems,

(2) a color coded simplified P&ID showing piping and component ultimate pressure capabilities, clearly identifying the interface junctions.

For all interfacing systems and components which do not meet the full RCS ultimate rupture strength criteria, justify, for each case, why it is not practicable to reduce the pressure challenge any further. This justification must be based upon engineering feasibility analysis and not solely risk benefit trade-offs.

For those interfaces where acceptable justification on the impracticability of full RCS pressure capability has been provided, there must be a demonstration of compensating isolation capability. For example, it should be demonstrated for each interface that the degree and quality of isolation

Question 440.45 (continued)

or reduced severity of the potential pressure challenges compensate for and justify the safety of the low pressure interfacing system or component. Adequacy of pressure relief and piping of relief back to primary containment are possible considerations. As identified in SECY 90-016 each of these high to low pressure interfaces must also include the following protection measures:

(1) the capability for leak testing of the pressure isolation valves (PIVs).

(2) valve position indication that is available in the control room when isolation valve operators are deenergized, and

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure systems and both isolation valves are closed.

Response 440.45

Combustion Engineering has reviewed the Intersystem LOCA (ISL) issue as defined by Generic Safety Issue 105 in NUREG-0933 and NUREG/CR-5102. Design changes based on PRA evaluations of ISL and engineering judgment have reduced the contribution of ISL to the core damage event frequency for the System 80+ design to approximately 1.0E-9 compared to the EPRI overall core damage frequency goal of 1.0E-5 as demonstrated in the System 80+ PRA.

Analyses of previous plants have identified the most significant potential ISL paths to be the Shutdown Cooling System (SCS) suction lines and the Low Pressure Safety Injection System injection lines. The design pressure of the SCS has been increased from 650 psig to 900 psig in the System 80+ design. The ultimate strength of the piping material will not, therefore be exceeded even if the SCS is subjected to normal RCS operating pressure. The design pressure of the SCS piping conforms to the EPRI requirements in Volume II, Section 5.2.3.2 of the ALWR Utility Requirements Document. Isolation provisions for the SCS which further reduce the possibility of ISL are discussed below.

The System 80+ Safety Injection System (SIS) design does not include a low pressure injection subsystem, thereby eliminating the other potential ISL path shown to be significant by evaluations of earlier designs. The design pressure of the SIS pumps and the injection piping from the discharge of the pumps to the outside containment isolation valve in each train is 2050 psig. The piping in these portions of the SIS can withstand normal RCS operating pressure. The design pressure of the injection piping from (and

Response 440.45 (continued)

including) the outside containment isolation valve to the RCS is equal to RCS design pressure. The SIS suction piping from the In-containment Refueling Water Storage Tank (IRWST) is designed to lower pressure in accordance with EPRI criteria in the ALWR Utility Requirements Document, Volume II, Section 5.4.3.2. Numerous valves isolate the SIS from the RCS as described below.

In addition to the SCS and SIS design changes, the Chemical and Volume Control System (CVCS) design has been revised to reduce the possibility of ISL. The Letdown Heat Exchanger is now located inside containment and its tube-side design pressure has been increased from 650 psig to 2485 psig. CVCS isolation and overpressure protection are discussed further below. The Process Sampling System (PSS) also interfaces with the RCS. A discussion of the isolation provisions and other features that address ISL for the PSS is also provided below.

Leak testing of pressure isolation valves is described in the response to NRC Question 210.88.

PSS

The design pressure of PSS piping that interfaces with the RCS is 2485 psig. In addition, flow restricting devices are provided in the RCS nozzle for each line to limit flow from a postulated downstream break to a value that can be accommodated by a charging pump. Two containment isolation valves are provided in each sample line that interfaces with the RCS. The normal position for these valves is closed. These valves are operable from the control room and have position indication in the control room. There is no leak detection instrumentation in these lines.

CVCS

This discussion refers to CESSAR-DC Figure 9.3.4-1, Sheets 1 and 2. Sheet 1 has been marked up to reflect revisions which will be incorporated into the next Amendment to CESSAR-DC. This sheet is attached for information. Sheet 2 can be found in CESSAR-DC. The P&ID coordinates for the valves listed below are provided in Table 440.45-1 of this response.

Letdown Line -- The letdown line is designed to RCS design pressure up to and including the letdown control valve isolation valves (CH-349 and CH-350). There are four valves in series upstream of each letdown control valve isolation valve. Two are inside containment (CH-515 and CH-516) and two are outside containment (CH-523 and CH-110P or CH-110Q). All valves upstream of the letdown control valve isolation valves can be operated from the control room, and control room position indication is provided for each.

Response 440.45 (continued)

These valves are normally open when the CVCS is operational (in Modes 1, 2, 3, 4, and intermittently, 5). Intermittently during Mode 5, and throughout Mode 6, letdown is isolated by closing valves CH-515 and CH-516. Since RCS pressure is reduced, any leakage past these valves, and a subsequent downstream pressurization, is negligible.

For any abnormal operational occurrence necessitating letdown isolation while the RCS is at full pressure (such as a CIAS, an SIAS, or a letdown line component malfunction), an RCS pressure challenge to lower pressure piping beyond CH-349 or CH-350 is obviated due to the extent of instrumentation, controls, and valves which ensure isolation well upstream of the lower pressure piping.

The letdown orifices (Sheet 1 of Figure 9.3.4-1, coordinates F-7) limit letdown flow to its maximum allowable value if the letdown control valves are fully open. The orifices are located in containment, upstream of the outer containment isolation valve (CH-523) in piping designed to RCS design pressure. The letdown line relief valve (CH-354) is located downstream of the letdown control valve isolation valves. CH-354 has a capacity equal to the capacity of the letdown orifices with the letdown control valve fully open. Overpressurization protection is thus provided for portions of the letdown flowpath designed to a pressure less than design pressure.

Charging Line -- The charging line design pressure equals or exceeds the design pressure of the RCS from (and including) the charging pumps, to the RCS. The line contains two check valves in series outside containment (CH-719 for pump 1, CH-705 for pump 2, and CH-639). The line also contains three check valves in series inside containment (CH-747, CH-433, and CH-448), along with two valves operable from the control room - one inside and one outside containment (CH-524 and CH-208). Position indication is provided in the control room for these two valves. Five check valves in series make it implausible that the charging pump suction piping could be overpressurized by the RCS.

Auxiliary Spray Line -- The auxiliary spray line (at coordinates H-7 and H-6 on Sheet 1 of Figure 9.3.4-1) is a path parallel to the charging line. It has the same piping design pressure rating and the same design configuration as the charging line. Five check valves in series separate the charging pump suction piping from RCS pressure. As stated above, it is implausible that the charging pump suction piping could be overpressurized by the RCS.

Seal Injection -- The design pressure of the reactor coolant pump (RCP) seal injection line is equal to or greater than the design pressure of the RCS from (and including) the charging pump to each RCP. There are four check valves in the line going to each RCP. Three are inside containment

Response 440.45 (continued)

and one is outside containment. For charging pump 1 to RCP 1A, for example, these valves are CH-719, CH-835, CH-787, and CH-866. The seal injection line upstream of the RCP header also contains a valve operable from the control room located outside containment (CH-255) and a valve operable from the control room inside containment for each RCP (CH-241, 242, 243, 244). Position indication is provided in the control room for these valves. The four check valves in series provide adequate protection against RCS overpressurization back to the charging pump suction piping.

Seal Bleedoff -- RCP seal bleedoff (seal injection return flow) is routed to the volume control tank. The design pressure of the bleedoff piping is equal to RCS design pressure from each RCP out to the first manual valve outside containment (CH-198).

Each of the four bleedoff lines (one from each RCP) contains an orifice and a valve (RC-430, 431, 432, 433; see CESSAR-DC Figure 5.1.2-2) operable from the control room. The orifice and valve are in piping with a design pressure equal to RCS design pressure. The orifice limits the controlled bleedoff flowrate from a postulated downstream pipe rupture to a value within the makeup capacity of a charging pump. In addition, the action of the RCP seals themselves restricts flow through a postulated break. The valve in each bleedoff line has position indication in the control room. Flow instrumentation in each line activates a high flow alarm in the control room once reaching the high setpoint.

Two valves operable from the control room are provided in the bleedoff line at the containment penetration (CH-506 is inside containment and CH-505 is outside containment). As noted above, the design pressure of this line is equal to RCS design pressure beyond CH-506. Position indication is provided for these valves in the control room.

SIS

The RCS/SIS and RCS/SCS interfaces referred to below are shown in CESSAR-DC Figures 6.3.2-1A through 6.3.2-1C. The P&ID coordinates for the valves listed below are provided in Table 440.45-1 of this response.

The design pressure of each RCS direct vessel injection line from the reactor vessel up to and including a motor operated isolation valve (SI-616 series, SI-602, 603) outside containment is equal to RCS design pressure. Each vessel injection line contains three check valves (SI-113 series, SI-217 series, SI-540 series) in series inside the containment in addition to the remotely actuated motor operated valve outside. The motor operated valve is operable from the control room and has position indication in the control room. Leakage past the check valve nearest the reactor vessel injection nozzle would actuate a high pressure alarm in the control room when pressure reaches the setpoint.

Response 440.45 (continued)

The design pressure of each hot leg injection line from the RCS up to and including a motor operated valve (SI-321, 331) outside containment is equal to RCS design pressure. Each hot leg injection line contains two check valves (SI-522, 523, 532, 533) in series inside the containment in addition to the motor operated valve outside. The motor operated valve can be operated from the control room and has position indication in the control room. Leakage past the check valve nearest the RCS would actuate a high pressure alarm in the control room when pressure reaches the setpoint.

The design pressure of the SIS from the SIS pump discharge to the motor operated valves outside containment in the vessel injection and hot leg injection lines is 2050 psig. The ultimate rupture strength of the piping in these lines can withstand normal RCS operating pressure.

A check valve (SI-434, 446) is located between each SIS pump and the point at which the direct vessel injection and hot leg injection piping branch. This valve provides additional isolation of the SIS pump suction piping from postulated application of high pressure via either the vessel injection or hot leg injection lines.

The direct vessel injection and hot leg injection piping interfaces with the Safety Injection Tank (SIT) fill and drain header in several locations. At each junction, the fill and drain piping is isolated from the RCS by two valves, i.e., a check valve (SI-217 series, SI-522, 532) inboard (closer to the RCS) of a normally closed, manually operated valve (SI-618 series, SI-322, 332) that can be operated from the control room. The manually operated valve is provided with position indication in the control room. Leakage past the check valve would actuate a high pressure alarm in the control room when pressure reaches the setpoint. The design pressure of the piping from the RCS up to and including the manually operated valve at each junction is equal to RCS design pressure. Fill and drain piping outboard of the manually operated valves (SI-661, 670, 682) has a design pressure of 2050 psig which would withstand full RCS pressure. The 2050 psig piping ultimately transitions to piping of low design pressure. A normally closed, manually operated valve is provided at each point of transition. The design pressure of these valves is 2050 psig; they are operable from the control room and they have position indication in the control room. The 2050 psig piping and the transition points to piping of low design pressure are inside containment. A postulated ISL in the low pressure piping would not, therefore, exit the containment.

The SIT's are isolated from the RCS by two check valves (SI-215 series, SI-217 series) in series during normal operation. The design pressure of the piping is equal to RCS design pressure from the RCS up to and including the second check valve. Leakage past the first check valve from the RCS would be indicated by a high pressure alarm in the control room when pressure reaches the setpoint. SIT high level and pressure alarms room would be

Response 440.45 (continued)

actuated if leakage into the SIT's caused their setpoints to be reached. The design pressure of the piping outboard of the second check valve from the RCS is 700 psig. The same design pressure is also employed for the SIT itself and for connected piping. Any postulated ISL at the SIT or connected piping would occur inside the containment.

SCS

The SCS suction piping is designed to RCS design pressure from the RCS up to and including the second of two motor operated isolation valves (SI-651, 652, 653, 654) in series inside the containment. These valves are closed during Modes 1, 2 and 3. In addition, there is also a motor operated valve (SI-655, 656) outside the containment. All three valves can be operated from the control room and have position indication in the control room. An alarm exists to notify the operator if the two motor operated valves inside containment are not fully closed coincident with the high RCS pressure.

A high capacity relief valve (SI-179, 189) provided for LTOP purposes is located downstream of the two motor operated valves inside containment in each SCS suction line. These relief valves would limit the effects of postulated leakage past the two upstream motor operated valves. The relief valves discharge to the in-containment holdup volume.

The design pressure of the SCS discharge piping is equal to RCS design pressure from the RCS up to and including a motor operated valve (SI-600, 601) outside containment. The motor operated valve is closed in Modes 1, 2 and 3. It can be operated from the control room and has position indication provided in the control room. In addition, there are four check valves (one outside containment, SI-168, 178 and three inside, SI-113 series, SI-217 series, and SI-540 series) in series with the motor operated valve. Leakage past the check valve nearest the RCS would actuate a high pressure alarm in the control room when pressure reaches the setpoint. The capability also exists to check for leakage across all five valves.

The design pressure of the remainder of the SCS is 900 psig. The ultimate rupture strength of the piping is sufficient to withstand normal RCS operating pressure.

TABLE 440.45-1 (SHEET 1)

INTERSYSTEM LOCA ISOLATION VALVE COORDINATE LOCATION

CVCS VALVES

FIGURE 9.3.4-1 SHEET 1		FIGURE 9.3.4-1 SHEET 2	
Valve Tag Number	Coordinate Location	Valve Tag Number	Coordinate Location
CH-110P, 110Q	E-6, E-6	CH-198	F-7
CH-208	G-7	CH-505	F-7
CH-241, 242, 243, 244	H-2, G-2 F-2, E-2	CH-506	F-7
CH-255	G-3	CH-705	E-2
CH-349, 350	E-6, E-6	CH-719	C-2
CH-354	D-6		
CH-433	H-6		
CH-448	H-6		
CH-515, 516	H-8, H-8		
CH-523	E-7		
CH-524	B-8		
CH-639	B-7		
CH-747	F-8		
CH-787	H-1		
CH-835	G-2		
CH-866	H-1		

TABLE 440.45-1 (SHEET 2)

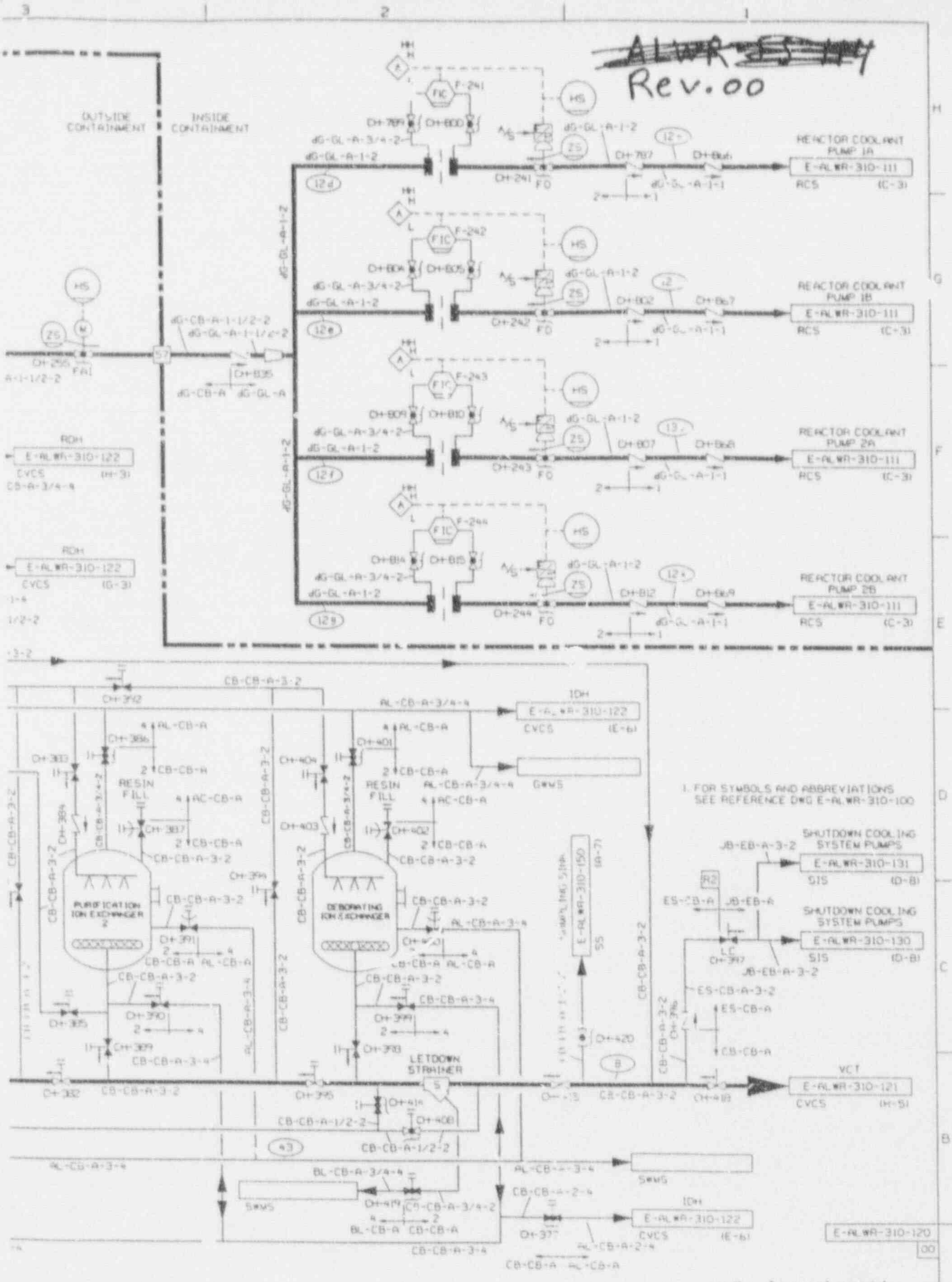
INTERSYSTEM LOCA ISOLATION VALVE COORDINATE LOCATION

SIS-SCS VALVES

FIGURE 6.3.2-1A		FIGURE 6.3.2-1B	
Valve Tag Number	Coordinate Location	Valve Tag Number	Coordinate Location
SI-434	B-4	SI-446	B-4

FIGURE 6.3.2-1C

Valve Tag Number	Coordinate Location
SI-113, 123, 133, 143	F-7, F-6, F-3, F-1
SI-168, 178	F-7, G-3
SI-179, 189	F-2, F-5
SI-215, 225, 235, 245	B-8, B-6, B-4, B-3
SI-217, 227, 237, 247	A-7, A-6, A-3, A-2
SI-321, 331	G-1, G-5
SI-322, 332	E-1, E-5
SI-522, 523, 532, 533	C-1, F-1, C-5, F-5
SI-540, 541, 542, 543	C-7, C-6, C-3, C-2
SI-600, 601	G-7, G-3
SI-602, 603	G-6, G-2
SI-616, 626, 636, 646	G-7, G-6, G-3, G-2
SI-618, 628, 638, 648	B-8, B-7, B-4, B-3
SI-651, 652, 653, 654	D-2, D-6, E-2, E-6
SI-655, 656	F-2, F-6
SI-661	C-1
SI-670	B-1
SI-682	C-1



440.45

Amendment I
December 21, 1990

CHEMICAL & VOLUME CONTROL SYSTEM
PIPING AND INSTRUMENTATION DIAGRAM

Figure
9.3.4-1

QUESTION 440.52

Per the staff position of BTP RSB 5-1, confirm that a boron mixing and natural circulation cooldown test will be performed in the first plant with a System 80+ design.

RESPONSE 440.52

Testing to verify adequate natural circulation and boron mixing was successfully conducted for the System 80 design at Palo Verde. The natural circulation cooldown capacity of the System 80+ design was evaluated in developing a response to RAI 440.51. The response to 440.51 indicates that the results of the System 80 natural circulation cooldown analysis apply to the System 80+ design in a conservative manner; that is, the results of the cooldown simulation for System 80 bound the System 80+ design. Based on the results of the Palo Verde testing and the evaluation of natural circulation cooldown capabilities that was performed for the System 80+ design, natural circulation cooldown testing of the System 80+ design is not considered necessary.

Since system 80+ differs from System 80 because of the direct vessel injection feature, a boron mixing test under natural circulation will be performed in the first plant with a System 80+ design. However, a cooldown is not considered necessary to confirm boron mixing requirements.

Question 440.72

Discuss the design criteria for the safety injection pumps, containment spray pumps, and the shutdown cooling pumps, and discuss whether the pump design criteria includes pump operations at or near shutoff head conditions?

Response 440.72

The functions and overall design criteria for the safety injection pumps are discussed in sections 6.3.1.1, 6.3.1.2.1 and 6.3.2.2.3. In addition, the design criteria for the safety injection pumps are that they must...

(a) provide sufficient flow to the RCS, following depletion of the Safety Injection Tanks (SIT), to keep the core adequately cooled following all loss of coolant accidents (LOCA),

(b) match the loss in RCS inventory from boiling due to decay heat beginning at about 20 minutes following a large break LOCA (LBLOCA), and

(c) inject water into the RCS during the feed portion of the feed-and-bleed operation of the Safety Depressurization System (SDS) for the purposes of removing decay heat from the core.

The pump head characteristics, in particular the shutoff and runout points, are selected to satisfy criteria (a) through (c) above, and those discussed in sections 6.3.1.1, 6.3.1.2.1 and 6.3.2.2.3. The safety injection pumps will operate at or near their shutoff points during certain main steam line breaks (MSLB) when the RCS pressure is at or near a value corresponding to the shutoff head of the pumps. The minimum flow recirculation (mini-flow) lines are designed to allow sufficient recirculation flow through the pumps so that they can operate at these conditions without damage. Mini-flow is directed to the IRWST and is available during all operating modes of the pumps.

The functions and overall design criteria for the containment spray pumps are discussed in sections 6.5.1.1, 6.5.1.2, and 6.5.2.2.1. In general, these design criteria were selected to be consistent with equipment previously licensed in System 80 designs.

The containment spray pumps are not expected to normally operate near their shutoff conditions. Nevertheless, mini-flow lines are provided for each pump, with heat exchangers, to prevent pump deadhead operation. The mini-flow lines are designed to allow sufficient flow to be produced by the pumps so that they can operate at these conditions without damage.

The functions and overall design criteria for the shutdown cooling pumps are discussed in sections 5.4.7.1.1, 5.4.7.1.2, and 5.4.7.2.2(E). In general, these design criteria were selected to be consistent with equipment previously licensed in System 80 designs. In addition, the shutdown cooling pumps are designed to produce flow to sufficiently remove decay heat using the shutdown cooling heat exchangers to limit the temperature rise across the core. This ensures that the RCS pressure does not rise above the maximum operating pressure for the SCS.

The shutdown cooling pumps are not expected to normally operate near their shutoff conditions. Nevertheless, mini-flow lines are provided for each pump, with heat exchangers, to prevent pump deadhead operation. The mini-flow lines are designed to allow sufficient flow to be produced by the pumps so that they can operate at these conditions without damage.

NRC Question 440.73

Provide an analysis for the potential for pump-to-pump interaction resulting in a pump dead heading scenario for the safety injection system, the containment spray and the shutdown cooling system. This analysis should identify all pumps and piping configurations that are pathways for pump-to-pump interactions including all shared common minimum flow recirculation lines and test lines (Reference NRC Information Notice 90-61, September 20, 1990).

Response

The design of the Safety Injection System (SIS), Containment Spray System (CSS) and Shutdown Cooling System (SCS) provides protection against pump "dead-head" operation resulting from pump-to-pump interaction. This has been accomplished by eliminating the need for low pressure safety injection pumps and by installation of an individual minimum recirculation flow (mini-flow) line for each pump.

In the System 80+ SIS, the flow rate required to be delivered to the RCS following large break loss of coolant accidents (LBLOCA) is provided by SIS pumps with suitable head curve characteristics. The low pressure SIS pumps of previous designs that had to produce both LOCA delivery and shutdown cooling flow rates have been eliminated in favor of dedicated pumps for safety injection and shutdown cooling functions. The result is the elimination of low pressure pumps (i.e., the System 80 low pressure safety injection pumps) connected (in their discharge) to higher pressure pumps that were required to operate near dead-head conditions during certain modes of operation. In System 80+, therefore, the source of pump-to-pump interaction that could cause pumps to operate at dead-head conditions has been eliminated.

The SIS, SCS and CSS designs also have individual mini-flow lines for each pump. The mini-flow connection is located immediately downstream of the pump discharge just upstream of the pump's discharge check and isolation valves. This eliminates the possibility of isolating the flow path to the mini-flow lines which would allow dead-heading of the pump if the main discharge flow path is closed. Furthermore, for the SCS and CSS, a dedicated loop around each pump is provided with a heat exchanger to remove pump heat in the event of a closed pump discharge path. These mini-flow lines do not have any remotely actuated valves. A locally operated manual valve that is provided to allow pump maintenance is locked open during all plant operating modes.

Finally, to further eliminate any pump-to-pump interaction, the general plant arrangement separates redundant trains of the SIS, SCS and CSS. The divisional boundary provides complete separation between divisions and effectively creates two identical support buildings. The result is a plant arrangement with two SI pumps, one SCS and one CSS pump located in each division. Within each division, the two trains are seperated by a quadrant wall and these trains are isolated from each other to the maximum extent practical. This precludes any cross connection to the remainder of these systems except through either the RCS or IRWST.

Question 440.82

Current editions of CESSAR-DC include Table 6.3.2-4a "SIS Flow Point Data-Injection Mode". This is the same table for the System 80 and the corresponding flow diagram does not have flow data points labeled for the location of the data readings. Does this table reflect flow point data for the System 80+ SI system?

Response

CESSAR-DC, Amendment I, does not include Table 6.3.2-4a, "SIS Flow Point Data-Injection Mode". This table, along with tables 6.3.2-4b, -4c, -4d and -4e were removed from CESSAR-DC in Amendment C. The tables were removed because System 80+ has a more simplified system operation in that the SISs performance is defined by one set of system operating characteristics. The System 80+ SIS provides for direct vessel injection where the discharge from each SI pump is piped directly to the reactor vessel. The splitting of flow from each (high pressure safety injection) pump and diverting it to all four injection nozzles on the cold legs as was done in System 80 has been eliminated. Furthermore, for long term cooling in System 80+, full flow from two of the four SI pumps is diverted to the hot leg. The requirement to obtain a 50%/50% hot leg/cold leg balance of flow from both HPSIP's has been eliminated.

Consequently, the SIS flow, whether for short term or long term cooling, will be identical as defined by the delivery curve. This is provided in table 6.3.3.3-1 of CESSAR-DC. Therefore, table 6.3.2-4a is not necessary for the System 80+ SIS design.

Question 440.110

Requirements for and analysis of safety injection systems (SIS) generally assume relatively short periods for operation of the SIS, on the order of several hours, up to perhaps one day. It must be recognized, however, that decay heat removal must continue to be provided after this initial period has passed, possibly for days, weeks, or even months. Under such circumstances, questions of reliability and maintainability become important. The staff is concerned that very-long-term-post-LOCA cooling is not being adequately considered in the design of SIS's, and is evaluating how such cooling might be incorporated into advanced reactor designs. The discussion in Sections 6.3 and 15.6 should be expanded and clarified to address the following items.

(1) Identify how the decay heat is transported to the ultimate heat sink. Include in this discussion the potential for cross connects between heat removal components that may improve overall system reliability.

(2) Identify what equipment is necessary for long-term post-LOCA cooling, and what the projected mission times are for the required equipment over the spectrum of accidents analyzed. Justify the mission times assumed.

(3) Where non-safety-related equipment is identified for use in long-term cooling, what reliability criteria should be assumed in determining the availability of this equipment?

(4) In the event of severe fuel damage to part of the core, considerable activity, and possibly fuel debris, may be transported into the SIS, with deleterious effects on system components. How will maintenance or repair be performed in a potentially high-radiation environment?

(5) Even without fuel damage, for long mission times, there is the possibility that key components, e.g., pumps and heat exchangers, will require maintenance and/or repair. How is this accommodated in the SIS requirements and in the long-term cooling plan?

(6) Has the necessity for very-long-term Post-LOCA decay heat removal been considered in your PRA? If not, why is this omission appropriate?

Response 440.110

(1) Long term decay heat removal is performed in one of two ways, depending on the size of the break.

For a small break LOCA (SBLOCA), RCS pressure and inventory control can be recovered within several hours to allow entry into shutdown cooling. Cooldown and depressurization of the RCS to shutdown cooling entry conditions is accomplished by using the steam generators and auxiliary pressurizer spray or the Reactor Coolant Gas Vent System (RCGVS), respectively. Once in shutdown cooling, decay heat is transferred to the Component Cooling Water System (CCWS) via the SCS heat exchangers. Heat exchangers in the CCWS then provide for the transfer of decay heat to the Station Service Water System (SSWS). Decay heat contained in the SSWS water is removed by the ultimate heat sink (pond, river, ocean, etc.).

For a large-break LOCA (LBLOCA), the RCS pressure may not be controllable and RCS inventory may be insufficient to allow entry into shutdown cooling. Under these conditions, simultaneous hot leg and direct vessel injection (DVI) will be initiated to maintain core inventory and flush the core to prevent boron precipitation. The safety injection pumps will take suction from the IRWST and will inject water into the hot legs and DVI nozzles. Water spilling out the break is directed to the Holdup Volume Tank (HVT), which replenishes the IRWST inventory once the HVT water level reaches the IRWST spillway elevation. Decay heat is removed from the core by either water boiling in the reactor vessel or water spilling out the break. Decay heat accumulated in the containment atmosphere due to boiling will be removed by the CSS and transferred to the IRWST. Water spilling out the break will eventually arrive in the IRWST through the the IRWST spillway from the HVT. Decay heat accumulated in the IRWST will be removed by the CSS or SCS since IRWST water is pumped through the containment spray or the shutdown cooling heat exchanger before being returned to the IRWST. Decay heat removed by the CCWS in these heat exchangers is transferred to the ultimate heat sink through the SSWS as described above in the SBLOCA discussion.

The reliability of long term cooling is increased by providing cross-connects which allow the interchangeable use of the shutdown cooling pumps and heat exchangers and the containment spray pumps and heat exchangers. The SCS and CSS pumps are identical which facilitates the use of these pumps for the interchangeable service.

(2) The following SIS equipment, and support systems equipment, is used in long-term post-LOCA cooling.

Safety Injection System

The IRWST, HVT, SI pumps, Safety Injection Tanks (SIT's), and associated valves and piping.

Shutdown Cooling System

SCS heat exchangers, pumps, control valves, relief valves, and associated piping.

Component Cooling Water System

CCWS pumps, heat exchangers, surge tanks, chemical addition tanks, radiation monitors, valves, and associated piping.

Station Service Water System

SSWS pumps and pump structures, pump structure screens, strainers, radiation monitors, valves, and associated piping.

Containment Spray System

CSS pumps, heat exchangers, and associated valves and piping.

The mission time requirements for equipment to remain in-service following a LOCA will meet or exceed the mission time requirements for previously licensed equipment in System 80 designs. More information regarding mission times is discussed in the response to RAI No. 270.2.

System 80+ incorporates a number of changes from the System 80 design that greatly improves the reliability for operation during accident recovery periods.

The SIS consists of four redundant mechanical trains, each with its own suction line from the IRWST, its own pump, and its own discharge line to the RCS. For breaks larger than the size of an injection line, each train, in conjunction with the SIT's, provides 50 percent of the minimum injection flow rate required to satisfy all LOCA performance requirements. For breaks equal to or smaller than the size of an injection line, each train provides 100 percent of the flow required to satisfy the LOCA performance requirements.

Direct vessel injection is used rather than cold leg injection to permit each of the SIS pumps to be sized for one-half of the capacity required for a cold leg break. Direct vessel injection (DVI) in conjunction with the use of four SI pumps increases the reliability of the SIS during LOCA events by maintaining the clear separation of the four SI loops and minimizing the number of valves within the system.

The System 80+ SIS pumps take their suction from the IRWST. The IRWST is connected directly to the Holdup Volume Tank, which serves as the "containment sump", via passive spillways. Therefore, there is no distinction between the injection and recirculation phases of SIS operation. By eliminating the need to realign the SI pumps from an outside Refueling Water Storage Tank to the containment sump, the reliability of the system is improved.

Two redundant SCS trains are available for small break, post-LOCA cooling. Each train has 100 percent capacity to ensure that one train will meet all SCS performance requirements. If both SCS trains become unavailable, the system design permits the CSS pumps to be aligned for long-term decay heat removal. Separation of the two SCS trains is readily maintained in the design.

In addition, the bleed function of the Safety Depressurization System has been incorporated into the System 80+ design to permit emergency decay heat removal.

(3) Satisfactory long-term post-LOCA cooling results can be demonstrated using only safety-related equipment. Control-grade equipment may be used according to plant procedures, but use of such equipment is not required or credited. Therefore, reliability criteria are not specified for design basis events. Control-grade equipment reliability is addressed for beyond design basis events as part of the PRA and Reliability Assurance Program.

(4) In the unlikely event of severe fuel damage to part of the core following a LOCA, fuel debris will not be transported to the SIS. The SIS circulates water from the IRWST to the reactor coolant loop. Provisions have been included in the IRWST to prevent transport of fuel debris or foreign matter into the system. All fluid directed to the IRWST passes through the HVT before entering the IRWST. Large debris carried by water flowing to the HVT will settle in that tank. Screens in the IRWST spillway inlets will prevent smaller debris from carrying over into the IRWST. Screens in the SIS suction inlets will prevent small debris (greater than 0.09 in. diameter) from entering the SIS. In addition, the SIS suction inlets are located above the bottom of the IRWST to prevent debris which has settled in the tank from being swept into the SIS suction lines.

The high radiation levels caused by the increased activity in the coolant will require special shielding to be installed should maintenance or repair of components be required under severe accident conditions.

(5) The need for maintenance and repair to SIS components during periods following a LOCA is minimized since (a) active components (such as pumps and valves), electric cabling, instrumentation and controls in the SIS are qualified to operate in a post-LOCA environment, and (b) redundant equipment is provided. If it is determined that equipment repair is needed on a very long-term bases, extraordinary measures would be developed at the time of the event, considering event specific conditions based on the best industry experience and knowledge available at that time.

(6) The PRA has considered the necessity of long-term post-LOCA decay heat removal consistent with the EPRI PRA Key Assumptions and Groundrules (EPRI ALWR Utility Requirements Document, Vol. II, Chapter 1, Appendix A).

QUESTION 440.111 (15.8)

Please provide a schedule for providing an ATWS analysis to demonstrate that the System 80+ ATWS response is within the bound considered by the staff during the deliberations leading to the ATWS Rule (10 CFR 50.62). This should include the analyses referenced on page B-92 which demonstrates that loss of feedwater with failure of turbine trip is the limiting peak pressure event.

RESPONSE 440.111

The "C-E NSSS Owner's Response to NUREG-0460, Volume 4 (CEN-134-NP) addressed the issue of the most limiting anticipated transient without scram (ATWS) event. The report concluded (see Section 2.2) that the total loss of main feedwater without turbine trip produces the highest primary pressures.

Appendix B of Section 3.1.12 of CESSAR-DC "Anticipated Transients Without Scram" will be revised as reflected in the attached markup to reflect the most recent ATWS analyses for System 80+. These analyses were performed on a best-estimate basis and demonstrated that the peak RCS pressures (i.e., cold leg) would not exceed 3140 psia for a moderator temperature coefficient of $-0.3 \times 10^{-4}\ \Delta\rho/^{0}F$ representing the most adverse expected MTC value for 99% of the fuel cycle.

RAI 440.111

3.1.12 ANTICIPATED TRANSIENTS WITHOUT SCRAM

3.1.12.1 ATWS Description

Anticipated Transient Without Scram (ATWS) is not an initiating event, but rather is a faulted response to an event requiring control element assemblies (CEAs) insertion for reactivity control. However, because of the significant impact that an ATWS has on plant responses, it is included as a separate initiating event class. The initiating event is defined to be the occurrence of a transient requiring reactor trip for reactivity control coupled with failure of a trip to occur due to either mechanical failure of the CEAs to insert or the failure of both the Reactor Protection System (RPS) and the Alternate Protection System (APS) to generate a trip signal. Because ATWS is included as a separate event, failure to trip was not addressed in the event tree for the other transient initiating event classes.

The ATWS is potentially a severe event in which the Reactor Coolant System goes through a pressure excursion due to a mismatch between the core heat generation rate and the Reactor Coolant System energy removal capability. Although 10 CFR 50.62[48] defines a prescriptive solution for the ATWS scenario in terms of prevention and mitigation, the success criteria for the event is given in NUREG-0460, Volume 3[49] and can be summarized as follows:

- For the Reactor Coolant System (RCS) pressures calculated, the integrity of the reactor coolant pressure boundary and the functionability of valves needed for long term cooling shall be demonstrated.

- The calculated radiological consequences shall be within the guidelines set forth in 10 CFR 100[50].

- The reactor fuel rods shall be shown to withstand the internal and external transient pressure so as to maintain a long term coolable geometry.

- The peak fuel enthalpy of the hottest fuel pellet shall not result in significant fuel melting.

- The probability of departure from nucleate boiling for the hot rod shall be shown to be low.

- The maximum cladding temperature and the extent of the $Zr\text{-}H_2O$ reaction shall be determined and shown not to result in significant cladding degradation.

 RAI 440.111

For the limiting ATWS scenario, the criteria relating to the pressure boundary integrity and functionability of the valves required for long term cooling are of primary interest. The concern is that if the peak pressure in the RCS exceeds Level C stress limits (approximately 3200 psia)[51], a breach of the primary coolant pressure boundary will occur and that the Safety Injection System check valves will be jammed closed. This would result in a LOCA with no RCS makeup available.

The course of an ATWS event is primarily dictated by a macroscopic energy balance on the Reactor Coolant System. Energy generated in the core and deposited in the coolant can be removed by various means; they are: the steam generators, the primary safety relief valves, and RCS leakage. Changes in the RCS pressure and temperature are produced as a result of an imbalance between the rates of energy deposition into and removal from the reactor coolant. All ATWS consequences are determined directly by the core power transient and the power imbalance transient. The relative consequences of ATWS events are thus determined by the relative magnitude of those plant parameters which govern these transients.

The energy generation within the core during the period of peak RCS pressure and maximum potential for clad damage is determined by the relative magnitude of Doppler and moderator temperature reactivity feedback. A power imbalance which produces an increase in moderator temperature and pressure coupled with a negative moderator temperature coefficient also produces a negative reactivity feedback which tends to reduce the core power and hence reduces the core power imbalance. During an ATWS event, primary coolant temperature increases. Since the assumed moderator temperature coefficient in the core is negative, the temperature increase results in an insertion of negative reactivity which reduces the core power. The moderator temperature coefficient will become more negative over the core cycle. Therefore, as the cycle progresses, the consequences of an ATWS event would become less severe, in that the core power reduction via moderator feedback will be greater, thus reducing the imbalance between the core heat generation rate and the RCS heat removal capability.

Insert (A) second page

~~Since RCS peak pressure and associated system stresses are the primary concerns during an ATWS, it has been determined by analysis that the complete loss of feedwater event with failure of turbine trip is the limiting at-power peak pressure event.~~

The loss of normal feedwater flow could result from a malfunction in the feedwater/condensate system or its control system. This

RAI 440.111

malfunction can be caused by a closure of all feedwater control valves, trip of all condensate pumps, or trip of all main feedwater pumps.

The loss of normal feedwater causes a reduction in feedwater flow to the steam generators when operating at power. This produces a reduction in the water inventory in the steam generators. Consequently, the secondary system can no longer remove the heat that is generated in the reactor core. Due to the assumed failure of the CEAs to insert on reactor trip, the core power remains at or near 100% of the initial level during the early part of the transient. The heat buildup in the primary system is indicated by rising RCS temperature and pressure, and by increasing pressurizer water level due to the insurge of expanding reactor coolant. The initiation of the ATWS event may be identified by means of the failure of CEA insertion on the reactor trip signal, sharp increases in RCS pressure and temperature, and a rise in steam generator pressure. The heat capacity of the primary and secondary coolant inventories, the discharge capability of the RCS and steam generator Safety and Atmospheric Dump Valves, and the action of the Emergency Feedwater System, Steam Bypass Control System, and the Chemical and Volume Control System all combine to provide the heat removal capability to limit the consequences of the reactor power generated during this incident.

Insert (B) next page

~~Realistic best estimate thermohydraulic analyses of a total Loss of Feedwater without Turbine Trip or Scram were run for MTCs of $-0.50\times10^{-4}\ \Delta\rho/^{\circ}F$ and $-0.30\times10^{-4}\ \Delta\rho/^{\circ}F$. The peak vessel pressures generated in these analyses were 3[illegible]89 psia for an MTC of $-0.30\times10^{-4}\ \Delta\rho/^{\circ}F$, and 2873 psia for an MTC of $-0.50\times10^{-4}\ \Delta\rho/^{\circ}F$. Therefore, since total Loss of Main Feed-water Flow without Turbine Trip is the limiting ATWS, an ATWS event will not exceed Level C stress limits for MTCs of $-0.30\times10^{-4}\ \Delta\rho/^{\circ}F$ or less.~~

Insert (C) next page

Figure B3.1.12-1 presents the core damage event tree for ATWS. The following subsections describe the individual elements on this event tree.

3.1.12.2 <u>ATWS Event Tree Elements</u>

3.1.12.2.1 ATWS Initiators

ATWS is defined to be an anticipated operational occurrence coupled with failure to insert negative reactivity via the CEAs. ATWS initiators, for this study, are defined to be all transients which tend to produce RCS pressure transients. These include Loss

RAI 440.111

INSERT A :

Since RCS peak pressure and associated system stresses are the primary concerns during an ATWS, it has been determined by analysis that the complete loss of feedwater event with failure of turbine trip is the limiting at-power event as documented in the "C-E NSSS Owner's Response to NUREG-0460, Volume 4 (CEN-134-NP)."

INSERT B :

The pressure continues to increase until the rate of RCS coolant expansion decreases due to the reduction in power caused by the core negative MTC. At this point, the PSV outflow matches, and then exceeds the surge line inflow to the pressurizer initiating a pressure decrease.

INSERT C :

Best estimate, thermal-hydraulic analyses of a total loss of feedwater without turbine trip or reactor scram were performed assuming the most adverse expected MTC value during 99% of the fuel cycle ($-0.30E-4\ \Delta\rho\ /^{\circ}F$).

The peak RCS pressure generated in this analysis was 3140 psia, which is below the level C stress limit of 3200 psia.

Question 440.115

Technical Specification 3.4.9 of CESSAR-DC Chapter 16 does not include the surveillance requirements for the demonstration of the emergency power supplies for the pressurizer heaters as proposed in the C-E Owners Group Standard Technical Specifications. Explain why.

Response 440.115

This surveillance requirement was inadvertently omitted from the System 80+ technical specifications. The response to RAI 430.23 defines pressurizer heater power availability as listed in Table 8.3.1-4.

The pressurizer heater power is supplied from a 4.16KV non-safety bus which may receive emergency power from the non-safety gas turbine or, if necessary, the diesel generator via a manual bus tie. A surveillance requirement will be added to the System 80+ Technical Specifications to demonstrate operability of an emergency power source for pressurizer heaters. This surveillance requirement will be included in a future amendment to Chapter 16.

QUESTION 440.116

Review of CESSAR-DC Section 6.8 on the In-containment Refueling Water Storage tank (IRWST) indicates that the Cavity Flooding System (CFS) is designed to "...flood the Reactor Cavity (RC) in the event of a severe accident for the purpose of covering core debris in the reactor cavity with water." Operation of the CFS during severe accident conditions requires manual actuation of several sets of motor operated valves (MOVs) in order to flood the RC.

These manual MOVs (valve Nos. SI-390 to SI 395) provide spillway links for water to flow from the IRWST to the Holdup Volume Tank (HVT) and then from the HVT to the reactor cavity. In light of the severe accident conditions in which these spillways would be used to fill the reactor cavity to ameliorate a curium-concrete reaction, explain why CE proposed to use an electrically dependent system requiring prompt operator action under stressfull conditions versus a passive system (such as one that employs a fusible metal plug for each HVT/RC spillway penetration) that will automatically open the spillways upon an elevated temperature produced by the corium. How will the timing of CFS operation be determined? What instrumentation will be relied upon? What criteria and timing will be employed to reduce the potential for a steam explosion due to dropping debris into a flooded cavity?

RESPONSE 440.116

For the System 80+ design the Reactor Cavity flooding is initiated using manual operator actions. These actions include manual actuation of motor-operated valves (MOVs) in order to provide spillway links for water to flow from the In-containment Refueling Water Storage tank (IRWST) to the Holdup Volume Tank (HVT) and then to the Reactor Cavity (RC).

Manual actions to flood the RC with IRWST fluid are predicated on indications of accident sequences potentially leading to the severe accident scenario, such as radiation alarms/indications in the containment, RCS and containment pressure and temperature indications/recordings, and reactor vessel level indication. Adequate power sources, such as batteries, would be available to provide power to facilitate the operation of the minimum set of instrumentation required even during a Station Blackout scenario.

In addition, sufficient time for appropriate operator actions is available during the severe accident scenario. Predictions by the MAAP code, which is employed to simulate the severe accident sequences, have indicated that reactor vessel failure occurs no sooner than 2 to 3 hours from the initiation of the accident. This suggests that adequate time is available for the operators to assimilate plant status information, properly diagnose the accident scenario, and take specific manual actions, sucn as opening of the MOVs for initiating reactor cavity flooding.

Manual actions also provide the flexibility to terminate cavity flooding should it be recognized later on that the transient would not lead to a severe accident scenario and that the plant can be stabilized using "conventional" Emergency Operating Procedures (EOPs).

A passive fusible plug for each spillway penetration could potentially delay timely cavity flooding till after vessel failure (since the temperature felt at the penetration may not be high enough to melt the plug). In addition, a fusible plug design would preclude testing of the cavity flooding system. Analytical studies have indicated significant quenching of the corium, adequate retention of core debris within the cavity, and scrubbing of the fission products, if the cavity is flooded prior to vessel failure. Cavity flooding prior to vessel faliure would also minmize the potential for any significant basemat melt-through. For these reasons, the System 80+ design uses an "on-demand" manual cavity flooding system for mitigating the consequences of a severe accident scenario.

The potential for a steam explosion causing damage to the reactor cavity and containment is considered to be very small. Following the reactor vessel failure during a severe accident, molten core debris would be released from the vessel into the reactor cavity. If water were accumulated in the cavity region prior to the vessel failure, molten debris-water interaction could be anticipated within the cavity. These were analyzed in references (1) and (2). As discussed in these references, the major influence of a potential steam explosion would be to disperse some of the water accumulated within the reactor cavity as well as further fragment and disperse molten debris that had been expelled from the reactor vessel at the time of the interaction. The evaluation of such events indicate that the energy yields would not be sufficient to threaten the integrity of either the reactor cavity or the containment boundary. Additional evaluations documented in References (3), (4), and (5) have confirmed this basic conclusion.

The evaluations and conclusions contained in the above references with regard to the potential for steam explosion are generally applicable to the System 80+ design because of the lower head mounted ICI design. This design would introduce corium into the cavity in a manner similar to that for the plants analyzed in the cited references.

Although the potential for steam explosion is minimal for the System 80+ design, the criteria and timing for operator actions for cavity flooding would account for this phenomenon. The specific operator guidance and instrumentation to be relied upon for manual cavity flooding would be developed as part of the overall accident management strategies for the System 80+ design. These strategies would be based on the NUMARC and NRC guidelines currently being developed.

References:

1. Zion Probabilistic Safety Study, Commonwealth Edison Company, September 1981.

2. Indian Point Probabilistic Safety Study, Consolidated Edison Company of New York and the Power Authority of the State of New York, April 1982.

3. "Steam Explosions in Light Water Reactors," Report of the Swedish Government Committee on Steam Explosions, Ds I 1981.

4. Probabilistic Risk Assessment, Limerick Generating Station, Philadelphia Electric Company, April 1982.

5. NUREG/CR-5567, BNL-NUREG-52234, "PWR Dry Containment Issue Characterization," Brookhaven National Laboratory, Prepared for the U.S. Nuclear Regulatory Commission, August 1990.

QUESTION 440.117

What severe accident analyses have been performed for establishing bases for the System 80+ CFS (IRWST/HVT/RC arrangement) that justify having only two HVT/RC spillways? What injection rates were assumed and is there enough total HVT/RC spillway discharge head to overcome pressures generated within the RC via the corium/concrete reaction (and steaming from initial water injection) based on a minimum IRWST water level under Technical Specification 3.5.4? Could corium/water interactions be violent enough to disperse core material and potentially block HVT/RC spillways?

RESPONSE 440.117

Severe accident analyses using the MAAP code were performed in support of the cavity flooding system (CFS) design for System 80+. These analyses helped in the overall design of the IRWST/HVT/RC arrangement. However, the choice of the number of spillways was based on sound engineering judgement in part to make it single failure-proof and to simplify the design.

The pressures generated within the Reactor Cavity due to steam produced by cooling of the molten corium are expected to be small. This is due to the fact that a relatively large opening between the Reactor cavity and the lower and upper compartments is present for relieving the steam produced by the cooling of the molten debris. Assuming about a 1 percent decay heat and a conservatively large steam production corresponding to twice the amount of the decay heat (to account for the cooling of the debris), the pressure drop between the Reactor Cavity and the opening is determined to be less than 0.04 psi. In comparison the hydrostatic head between the HVT water level and the location of the Reactor Cavity spillway entrance into the cavity is significantly higher (> 4 psi). This means that even with the conservatively large steam production from decay heat removal and debris cooling, the backpressure in the Reactor Cavity will not buildup significantly since all of the steam generated would be expelled into the containment with the buildup of a very small pressure difference. This allows for adequate delivery of IRWST fluid into the cavity.

A discussion of corium/water interactions leading to steam explosions is provided as part of the response to Q.440.116. This discussion indicates that the steam explosions would not yield sufficient energy to threaten the integrity of the cavity walls or the containment boundary. Based on the same discussion it can be concluded that the probability of blockage of both HVT/RC spillways due to the interaction of core debris with water would not be violent enough to potentially block <u>both</u> HVT/RC spillways with the debris.

Question 440.118:

(GSI-23: Reactor Coolant Pump Seal Failures)

The staff's draft safety evaluation report (DSER) for the EPRI Evolutionary Utility Requirements Document (URD) indicated that all new plant designs should provide independent RCP seal cooling for coping with station blackout (SBO) conditions. This measure was adopted, in part, since the EPRI URD specifies that the Chemical Volume and Control System (CVCS) is not required to perform safety functions and therefore maintains seal integrity with a non-safety grade seal injection (SI) system. The URD specifies that the CVCS design, reliability, and availability should be enhanced through design improvements, judicious location of components, and selected application of redundancy and diversity requirements.

Initial review of CESSAR-DC Section 9.3.4 on the CVCS design indicates that the CVCS is ". . . designed as a non-safety related system. . . In particular, the CVCS is not required to ensure the capability to prevent or mitigate the consequences of plant accidents." This statement is inconsistent with the ABB-CE proposed resolution on Generic Safety Issue (GSI) 23 "Reactor Coolant Pump (RCP) Seal Failures" (pages A-14 to A-16). The ABB-CE proposed technical resolution of GSI-23 relies solely upon the non-safety related CVCS seal injection to maintain RCP seal integrity and subsequently prevent a potentially core damaging seal LOCA.

With respect to the RCP seals, the CVCS clearly functions to support reactor coolant pressure boundary (RCPB) integrity. In addition, the System 80+ design has reduced the number of charging pumps from the three positive displacement pumps (PDPs) (two parallel trains and one common shared PDP) for the System 80 design to two (parallel trains) centrifugal charging pumps (CCPs). Even though CCPs tend to exhibit enhanced reliability characteristics compared to PDPs, the reduction in the number of charging pumps would superficially imply a reduction in the reliability of the system unless other engineering factors (as shown in a failure mode and effects analysis (FMEA) and fault tree analysis) are documented to support analysis for improved System 80+ CVCS reliability, especially in the context of GSI-23.

ABB-CE should provide a comparative CVCS reliability analysis of the System 80+ versus the System 80 to determine if the System 80+ CVCS design is consistent with the EPRI guidelines and that seal integrity, and consequently the reactor pressure boundary, is not compromised during normal plant operations. This analysis should be appropriately addressed in the relevant sections of CESSAR-DC.

Response 440.118:

In paragraph 1, the NRC reviewer states: ". . . the EPRI URD specifies that the Chemical and Volume Control System (CVCS) is not required to perform safety related functions . . ." To implement this position, the EPRI URD states that the entire CVCS can be designed as non-safety grade (i.e., non-nuclear safety (NNS)), and all safety grade functions performed by

Response 440.118 (Cont'd):

current generation systems can be transferred to other dedicated safety systems. We agree with, and have implemented the EPRI requirement relative to not crediting the CVCS with safety-related functions. We have however elected to take a different approach regarding the safety classification of piping and components.

In the System 80+ design, all CVCS safety functions have been transferred to other dedicated safety systems. This transfer involves safety functions which were previously credited to the CVCS, such as depressurization, and boron addition for reactivity control. However, the transfer of safety related functions to other dedicated systems has not resulted in a relaxation of CVCS design standards for reliability, redundancy, and availability (i.e., the System 80+ CVCS has not been designed as non-nuclear safety (NNS)). In accordance with current NRC regulatory criteria (Regulatory Guide 1.26), the System 80+ charging and letdown portions, including seal injection and reactor coolant pump bleedoff, are designed as Safety Class 3. Onsite alternative AC (AAC) power is provided to the charging pumps for their continued operation during a Station Blackout in accordance with draft Regulatory Guide 1008, which specifies design requirements for alternate RCP Seal Cooling systems. As described in CESSAR-DC Section 8.1.4.2, the installation and design of the alternate AC power source is in compliance with NRC Regulatory Guide 1.155, "Station Blackout". Consequently, the System 80+ CVCS provides two diverse electrical power sources for RCP seal cooling for ensured seal integrity.

In paragraph 2 of the RAI it is pointed out that there is an "inconsistency" between:

a) the CVCS design, since it is non-safety related, and

b) the C-E proposed resolution to Generic Safety Issue (GSI) 23, which relies on seal injection to maintain seal cooling.

Designating the CVCS as non-safety related has not diminished the quality of the design. As discussed in the paragraphs above, the system (in particular, the charging portion, including seal injection) is designed to ASME Section III, Safety Class 3 standards. The system receives normal power from redundant, non-safety related buses. For events where normal station power is available, the CVCS is operated to provide seal injection for seal cooling. For a Station Blackout event, the system receives power from the AAC bus, and seal injection flow is continued. Continued seal injection during this event provides the alternative to component cooling water for seal cooling flow, assuring seal integrity, and subsequently precluding a "potentially core damaging seal LOCA". The reactor coolant pressure boundary remains intact with continued seal cooling provided by the CVCS.

In paragraph 3 of the RAI, it is stated that the use of 2 centrifugal charging pumps is less reliable than 3 positive displacement pumps. On the surface, 2 pumps could suggest less redundancy than 3 pumps. However, understanding how 2 centrifugal pumps function in the System 80+ CVCS design provides assurance that there has been no compromise in reliability.

Response 440.118 (Cont'd):

Mechanical Design

In the previous System 80 design, each of three pumps delivered 44 gpm. Therefore, 132 gpm (maximum) of charging was possible. For the System 80+ design, however, one centrifugal pump can provide flow over the entire range of required CVCS flowrates (i.e., from 44 to 132 gpm). Consequently, the complete flowrange of all three positive displacement pumps is achieved with one centrifugal pump. The other centrifugal charging pump is a completely redundant, installed spare. With a single mechanical failure of one centrifugal charging pump, therefore, the other is available to provide the complete charging flowrange. For System 80, with the same failure, only 88 gpm maximum would be achievable.

The design with centrifugal charging pumps, therefore, exhibits enhanced reliability over the positive displacement pump design.

Electrical Design

In the positive displacement pump CVCS design, pump 1 is powered from bus A, pump 2 is powered from bus B, and pump 3 has the capability of being powered from either bus A or B. Upon a loss of one electrical bus, therefore, only two pumps can be operated, with a total flowrate of 88 gpm.

In the centrifugal pump CVCS design, pump 1 is powered from bus A and pumps 2 is powered from bus B. Loss of an electrical bus would result in the ability to provide the complete charging flowrange (44 to 132 gpm), a design enhancement over the positive displacement based CVCS design.

Consequently, the failure of a centrifugal charging pump due to either a mechanical or electrical failure would have no adverse system impact. Continued charging flow, over the entire flowrate range, is available from the installed spare (in the case of a mechanical pump failure) or from the pump on the bus which continues to receive electrical power.

The RAI has suggested that a Failure Modes and Effects Analysis and Fault-Tree Analyses be submitted for the CVCS. C-E believes that Failure Modes and Effects Analyses and Fault-Tree Analyses for non-safety related systems need not be reported in CESSAR-DC, although they have been performed during the System 80+ CVCS design process with acceptable results.

Summarizing, seal injection furnished by the CVCS is the best design and operational approach to protecting the seals during a station blackout and serves as a redundant, diverse system for seal cooling.

Question 440.119:

In addition to the discussion in the previous RAI (440.118) on GSI-23, the following information provides additional clarification on the staff's position concerning GS-23 relative to advanced reactor designs. Probabilistic risk assessment (PRA) analyses have indicated that the overall probability of core damage due to a small break LOCA could be dominated by RCP seal failures. RCP seal failures have been classified as LOCAs with RCS leakage up to 500 gpm per RCP. The primary objective for the resolution of GSI-23 includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and off-normal conditions.

RCP seal failure scenarios are separated into two categories:

(1) those resulting from mechanical-induced or maintenance-induced failures, and (2) those resulting from a loss of seal cooling such as SBO.

The first aspect of GSI-23 deals with seal failures during normal operation and have been demonstrated through numerous in-plant occurrences. Failures have occurred from maintenance errors, vibration, corrosion, plant transients, contamination, abnormal pressure staging, operator errors, improper venting, improper instrumentation, defective parts, and other causes. Normal condition seal LOCAs have resulted in unisolable RCS leakage at rates up to 500 gpm per RCP.

The second aspect of GSI-23 deals with a loss-of-seal cooling during off-normal conditions. Loss of seal cooling may occur under the following conditions:

(1) Loss of all AC power (i.e., SBO as defined by 10 CFR 50.2).

(2) Loss of component cooling water (CCW) function independent of SBO.

(3) Loss of service water (SW) function independent of SBO.

(4) Inadvertent termination of RCP seal cooling due to a safety-injection/containment-isolation signal or loss of a pneumatic system.

Seal injection availability during off-normal conditions is of particular concern for advanced reactor designs. Isolation of seal injection to RCPs has been identified as a significant contributor leading to high leakage (Ref. NUREG/CR-4948) for operating reactors. The probability for loss of seal injection may be exacerbated by a non-safety related CVCS for new designs. This is due to the fact that the non-safety CVCS would not be required to meet the single failure criteria (redundancy, diversity, electrical independence, etc...) or withstand a design basis accident (DBA), even though the CVCS is apparently composed of safety and seismically classified components. CVCS unavailability leaves no method of alternate seal injection and introduces the possibility of additional uncertainty in the capability to maintain seal integrity, and subsequently the RCPB. Also of concern to the staff, is the fact that seal injection following LOOP, is normally supported by the alternate AC (AAC) power system, rather than the emergency DGs.

Question 440.119 (Cont'd):

In the event that SBO conditions exist and AAC is not available as stated in CESSAR-DC GSI resolution, the staff questions your assumption that the shaft seals are capable of limiting leakage to a maximum of 8 gpm per pump without cooling. This is based on research findings for GSI-23, seal hydraulic-instability leading to seal faces "popping open," given a sufficient loss of inlet subcooling or seal back pressure.

Due to the above concerns, it does not appear that the CESSAR-DC resolution of GSI-23 adequately addresses the issues. Based upon recent GSI-23 research results, it appears that the following approach would provide more effective resolution of GSI-23 vulnerabilities. Please address the applicability and feasibility of implementing these criteria for CESSAR-DC.

(1) Treat the RCP seal assembly as an item performing a safety related function similar to other components of the reactor coolant pressure boundary, applying quality assurance requirements consistent with Appendix B of 10 CFR Part 50 and applicable General Design Criteria (GDC) of 10 CFR Part 50, Appendix A. This measure would bring the System 80+ RCS design closer to the intent of GDCs 14 and 30.

(2) RCP manufacturer recommended instrumentation and instructions for monitoring RCP seal performance should be provided on the use of monitored parameters for early degradation detection in order to prevent or mitigate a cascade failure of multi-stage seal assemblies. As a minimum, RCP seal procedures should be provided for normal plant operation conditions, including pump startup, pump shutdown, and off-normal conditions. Procedures for off-normal conditions should include loss of seal injection flow, loss of cooling to seal coolers (e.g., seal coolers, thermal barrier heat exchangers, etc...), loss of all seal cooling (consistent with Requirement No. 3 stated below), and pump restart after loss of all seal cooling events.

(3) Provide an independent seal cooling system which will be operable during off-normal plant conditions involving loss of all seal cooling events. This system should be seismically qualified and independent of the CVCS and associated support systems to the extent practicable. Some existing piping run may be shared, if the probability of failure is demonstrated to be acceptably low, or in the event of pipe failure the leak can be easily identified, isolated, and seal cooling maintained. The system should have appropriate Technical Specifications for Surveillance Requirements and Limiting Conditions for Operation.

RESPONSE 440.119:

In addition to the concerns stated in question 440.118, the NRC states that CESSAR-DC resolution of GSI-23 does not adequately address the issues. The issues are identified as RCP seal failure scenarios separated into two categories:

Response 440.119 (Cont'd):

(1) Those resulting from mechanical-induced or maintenance induced failures during normal operation.

(2) Those resulting from a loss of seal cooling such as station blackout (SBO).

Addressing the first category of seal failures, it is stated that numerous seal failures have occurred during normal operation. It is further stated that RCP seal failures have been classified as LOCA's with RCS leakage up to 500 gpm per RCP.

The 500 gpm seal failure occurred in the 1970's and resulted from continued operation of the RCP with damaged seals. This one worst case seal failure is not representative of the seal failure leakage rates which have occurred. In fact seal leakage rates due to seal malfunction have been considerably below the 25 gpm per pump criteria established in Regulatory Guide 1.155. Seal performance during normal operation has improved significantly since 1983 as stated in the NUMARC response, dated September 30, 1991 to Draft Regulatory Guide DG-1008 and as supplemented in the CEOG response, CEN-408, to DG-1008.

In the CEOG report (CEN-408) 10 utilities with 15 operating C-E designed plants were surveyed to determine their RCP seal operating experience since 1983. A seal assembly failure is defined as an occurrence when two or more seal stages are not functioning as designed. A failure may result in external leakage from the seals or excessive controlled bleed off flow which is contained and piped off to the volume control tank. A total of 23 failures were reported which required seal assembly replacement and met the above defined seal failure criteria. Of this amount, only three failures resulted in external leakage from the seals into the containment. The maximum external seal leakage was 3.0 gpm which is well below the 25 gpm criteria. The other 20 failures involved higher than allowed controlled bleed off flow which does not constitute external leakage from the RCP. The 23 failures span a 8 year time frame for 59 RCP's and are considered a reliability concern and not a safety concern by the industry.

It should be noted that NUREG-1401, Regulatory Analysis for Generic Issue 23, does not differentiate between those seal failures which resulted in external seal leakage from the RCP and those seal failures which caused a higher than acceptable controlled bleed off flow to the volume control tank or similar collection tank. Lumping these two different types of seal malfunctions together results in higher than actual external leakage seal failure rates and tends to present an inaccurate picture of actual industry wide seal performance.

The RCP seals to be used in the System 80+ plant are the same as those used in the Palo Verde plant. The performance of these multiple stage seals has been excellent at Palo Verde and no unplanned shutdowns from normal operation can be attributed to performance of the seals alone. There have been several incidences of seal malfunction during loss of seal cooling events, but the external seal leakage was well below the 25 gpm criteria. These incidences are included in the CEOG report, CEN-408, and additional information on these events is provided in C-E response to question 440.120.

Response 440.119 (Cont'd):

Addressing the second GSI-23 seal failure category which deals with seal performance during loss of seal cooling, the NRC lists the following conditions:

(1) Loss of all AC (i.e. SBO as defined by 10CFR50.2).

(2) Loss of Component Cooling Water (CCW) function independent of SBO.

(3) Loss of Service Water (SW) function independent of SBO.

(4) Inadvertent termination of RCP seal cooling due to a safety injection/containment-isolation signal or loss of a pneumatic system.

System 80+ RCP seal cooling is accomplished by two independent and redundant seal cooling systems: seal injection and component cooling water. Before addressing the above NRC defined conditions, a description of the System 80+ component cooling water system and service water system is provided in the following two paragraphs.

> The component cooling water system (CCWS) consists of two separate, independent, redundant, closed loop, safety related divisions. Either division of the CCWS is capable of supporting 100% of the cooling functions required for a safe reactor shutdown. A single failure of any component in the CCW system will not impair the ability of the CCW system to meet its functional requirements. Each division consists of an essential and non-essential cooling loop. The essential loops are composed of Safety Class 3 piping and components and cool safety related loads including the water cooled motors on the charging pumps. The non-essential loops are composed of non-nuclear safety piping and components and cool non-safety related loads such as the reactor coolant pumps.
>
> Cooling water for the safety grade CCWS pumps and heat exchangers is provided by the service water system (SWS). The SWS consists of two separate, redundant safety related divisions. Each division cools one of the two CCWS divisions. A single failure of any component in the SWS will not impair the ability of the SWS to meet its functional requirements.

The RCP's and supporting cooling systems are designed to withstand the NRC defined conditions as stated below:

(1) For the loss of all AC power (i.e., SBO) condition, the RCP seals are provided with seal cooling via an on-site alternate AC (AAC) power source which is used to power the charging pumps which supply seal injection (SI) water to cool the shaft seals. The AAC power is also used to power the CCW system pumps and SW system pumps to ensure component cooling water (CCW) is furnished to the charging pumps. The 10 minute delay mentioned in Regulatory Guide 1.155 for furnishing AAC power to the charging pumps, CCW pumps and SW pumps will not cause any problems for the RCP seals. The RCP seals are capable of withstanding without damage a loss of seal injection water and component cooling water for in excess of 10 minutes with the pumps in an idle condition as would happen during a loss of all AC power.

Response 440.119 (Cont'd):

(2) For the loss of non-essential component cooling water function independent of SBO, the shaft seals are furnished with seal injection (SI) water to cool the shaft seals. Essential CCW is furnished to the charging pumps. Since the essential CCW system is safety grade and meets the single failure criteria it is not credible that both divisions of the CCW system would be lost.

(3) Complete loss of the service water (SW) system is not credible since the two divisions are safety grade, fully redundant and meet the single failure criteria.

(4) The RCP seal cooling system is unaffected by a safety-injection actuation signal (SIAS) or a containment-isolation actuation signal (CIAS). The System 80+ RCP operational strategy has incorporated the guidance set forth in NRC Generic Letter 83-10a by including design provisions which preclude seal damage due to the loss of the component cooling water due to a SIAS or CIAS. CCW to the RCP's is not isolated on an SIAS or CIAS. Seal injection flow is not isolated on any PPS or ESFAS generated signal.

On a loss of air, CCW flow and seal injection flow to the RCP seals are unaffected. There are no pneumatically operated valves in the CCW flow path. Although there are pneumatic valves in the seal injection line, these valves fail in a position which ensures continued seal injection flow to the seals.

Improved seal cooling availability during off-normal conditions is a design basis of the System 80+ design. In response to the NRC concern that isolation of seal injection (SI) water to RCP's has been identified as a significant contributor leading to high seal leakage, the System 80+ RCP's have independent and redundant seal cooling via SI water and CCW and are capable of operating with SI water only or CCW only. During normal operation both the SI and CCW methods of seal cooling are in operation. This pump seal cooling capability is explained in more detail in our response to 440.125.

The probability of a loss of seal injection is not exacerbated by the System 80+ CVCS design. This issue is discussed in detail in the response to Question 440.118. Additionally, CVCS unavailability does not impact the CCW supply to the seals as the alternate method for seal cooling. The LOOP scenerio is not a concern, since the CVCS provides seal injection powered from the AAC power source. Simultaneously, CCW is provided to the seals, as this system is powered from the emergency diesel generators.

The NRC staff also questions the assumption that the shaft seals are capable of limiting seal leakage to a maximum of 8 gpm per pump in the unlikely event that all seal cooling is lost with the pump in an idle condition. As stated in our CESSAR DC response to GSI-23, this capability is based upon operating and test experience with multiple stage hydrodynamic shaft seals used in C-E designed plants. The capability is particularly based on the operating events at the Palo Verde plant. Additional information on these events is found in our response to 440.120.

Response 440.119 (Cont'd):

In the last paragraph of the subject question (440.119), the NRC repeats the concern that the CESSAR-DC resolution of the GSI-23 does not adequately address the issues. The NRC further requests that C-E address the applicability and feasibility of implementing the three resolutions proposed by Draft Regulatory Guide DG1008.

The first two DG1008 resolutions are summarized as follows:

(1) Treat the RCP seal assembly as a component of the safety related reactor coolant pressure boundary. Apply quality assurance requirements consistent with 10CFR50 Appendix B and applicable General Design Criteria of Appendix A to 10CFR50.

(2) Provide RCP manufacturer recommended instrumentation and instructions for monitoring seal performance and detecting incipient RCP seal failures. Provide RCP operating procedures to protect the seals for both normal and off-normal plant conditions.

The first resolution is not applicable to the System 80+ RCP seals because the seals are already designed and manufactured to a quality assurance program which complies with many of the 10CFR50 Appendix B requirements in order to provide the reliability demanded by the end user. In addition, each seal assembly receives final manufacturing processing in a clean room where temperature, humidity and airborne particulates are controlled. Dimensional requirements are verified by computer-aided measurement systems. Each seal assembly is hydrostatically pressure tested at 150% of RCS design pressure and operationally tested in a seal test rig which simulates actual pump operating conditions. The costs necessary to implement resolution (1) completely will not provide any additional improvement in seal performance.

The second resolution will be implemented for the System 80+ design based on using the successful and applicable instrumentation, instructions and operating procedures from the System 80 plant design as implemented at Palo Verde and any revisions provided by the pump supplier at the time of component procurement.

C-E's position on these two resolutions is consistent with the industry positions taken in the NUMARC responses to DG-1008 and as supplemented in the CEOG response, CEN-408.

The third DG1008 resolution calls for an independent seal cooling system which will be operable during off-normal plant conditions involving loss of all seal cooling systems. As previously stated in this response, the System 80+ design incorporates independent seal cooling via an on-site AAC power source which is used to power the charging pumps which provide seal injection to cool the RCP seals. Thus, the seal injection system meets all design requirements stated in Appendix A to DG 1008 for independent seal cooling systems.

Summarizing, the System 80+ RCP seals are a highly reliable multiple stage design capable of withstanding off-normal operating conditions as proven by operating experience at the Palo Verde plant. The seals are manufactured to

Response 440.119 (Cont'd):

high quality standards to ensure high reliability. The seals are cooled by two independent, diverse and redundant systems, i.e., seal injection and component cooling water. These systems are designed to provide seal cooling under various off-normal operating conditions, particularly station blackout, where an on-site alternate AC power source is used to power the charging pumps.

Question 440.120:

According to CESSAR-DC Appendix A proposed resolution of GSI-23, ABB/C-E cites an operating event at the Palo Verde Nuclear Generating Station (PVNGS) as partial demonstration of the CE-KSB pump seal capability during SBO conditions or loss of CCW (justifying no large leak rates following loss of seal injection). However, there is an inadequate discussion of the operational history of Palo Verde Unit No. 2 RCP 2B (seals) to properly support RCP seal integrity during SBO conditions or loss of seal injection. CESSAR-DC Appendix A (page A-15) describes the licensee event in the following excerpt:

"In April 1986, Palo Verde Unit 2 RCP 2B experienced a loss of CCW and SI for three hours. During this three hour interruption the pump was operated for 10 minutes before it was tripped. This resulted in the pump seals being exposed to RCS hot standby temperature conditions. No loss of seal function occurred and there was no measureable increase in the leakage to the containment. Following this event, the affected RCP was placed back into service without inspection of the seals. The RCP was operated for several months prior to a normal refueling and maintenance shutdown."

The staff does not believe that this isolated event provides adequate justification on seal performance following loss of SI. It should also be noted that the System 80+ GSI-23 resolution did not address a subsequent Palo Verde event that involved the same Unit No. 2 RCP 2B. LER 86-041 (dated 07/31/86) states that on July 1, 1986, the PVNGS Unit No. 2 developed an unidentified leak greater than 1 gallon per minute (gpm) in the reactor coolant system. A closer examination of the Palo Verde LER data base indicates a failure of the RCP 2B seals. The information submitted for GSI-23 resolution has not provided any information ruling out the possibility that the previous event may have contributed to the July 1 seal failure.

Please provide any additional operational data which you believe supports your belief that loss of RCP seal injection will not result in significant seal failure and resulting large loss of RCS coolant.

However, as stated previously, the significant uncertainties regarding seal failure modes and likelihood would prudently require that GSI-23 resolution include independent (SBO capable) seal cooling as discussed in RAI 440.119 (Item 3). The staff recommends that such an approach be utilized in demonstrating technical resolution of GSI-23.

Response 440.120:

In the subject question, it is stated that the CESSAR-DC Appendix A proposed resolution of GSI-23 does not provide adequate discussion of the operational history of the Palo Verde plant RCP's to support seal integrity during station blackout (SBO) conditions or loss of seal injection. An excerpt from CESSAR-DC Appendix A which describes an event at Palo Verde is cited as follows:

Response 440.120 (Cont'd):

"In April 1986, Palo Verde Unit 2 RCP 2B experienced a loss of CCW and SI for three hours. During this three hour interruption the pump was operated for 10 minutes before it was tripped. This resulted in the pump seals being exposed to RCS hot standby temperature conditions. No loss of seal function occurred and there was no measurable increase in the leakage to containment. Following this event, the affected RCP was placed back into service without inspection of the seals. The RCP was operated for several months prior to a normal refueling and maintenance shutdown."

It is also noted that the SYSTEM 80$^+$ GSI-23 resolution does not address a subsequent Palo Verde event that involved the same Unit No. 2 RCP 2B. This event is described in a Licensing Event Report with the following comments from the NRC staff:

"LER 86-041 (dated 07/31/86) states that on July 1, 1986, the PVNGS Unit No. 2 developed an unidentified leak greater than 1 gallon per minute (gpm) in the reactor coolant system. A close examination of the Palo Verde LER data base indicates a failure of the RCP 2B seals. The information submitted for GSI-23 resolution has not provided any information ruling out the possibility that the previous event may have contributed to the July 1 seal failure."

Additional operational data to support the position that "loss of RCP seal injection will not result in significant seal failure and resulting large loss of RCS coolant" has been requested. It should be noted that the RCP seals have redundant seal cooling methods and that the seals are unaffected by a loss of seal injection water provided component cooling water is available. Therefore, it is believed that the NRC reviewer intended to request additional operational data for a loss of both seal injection and component cooling water.

C-E has reviewed the available information from the April 1986 and July 1986 events at Palo Verde and offers the following additional information. A review of LER 86-015 which describes the April 1986 event and LER 86-041 which describes the July 1986 event shows that the earlier event was a contributor to the July 1, 1986 seal malfunction.

LER 86-015 states that seal injection (SI) water was shut off to all four pumps in Unit No. 2 because of temperature control problems with the seal injection heat exchanger which heats SI water if the water temperature drops below a certain value. SI water was restored to three of the RCP's, but not RCP2B because of an apparent plugged filter in the pump cooling circuit. RCP2B was shut down, the filter flushed and normal SI water was restored to RCP2B. LER 86-015 shows that RCP2B was without SI water for approximately 3 hours and although not indicated in the LER, component cooling water was shut off to RCP2B for all or part of the 3 hour period. The seals in RCP2B were subjected to temperature transients during the event with the highest recorded temperature approaching 250°F. The LER 86-015 event is the same as the April 1986 event described in the CESSAR-DC Appendix A resolution of GSI-23.

Response 440.120 (Cont'd):

LER 86-041 indicated that on July 1, 1986 the leakage from Unit No. 2 RCP2B exceeded 1.0 gpm and the plant was shutdown at which point it was decided to replace the seals in all four pumps. Our records indicate that the leakage from RCP2B was between 2 and 3 gpm; considerably below the 25 gpm per pump criteria of Regulatory Guide 1.155. Subsequent examination of the seals did not reveal any evidence of the seal "popping open" phenomenon described in Draft Regulatory Guide DG 1008.

Evaluation of these two LER's indicates that the RCP seals were subjected to an off normal event (April 1986); stabilized after normal conditions were reestablished and continued to operate for three more months before RCP2B exhibited a leak considerably below the 25 gpm criteria. The total operating time for the RCP2B seals was approximately 14 months before replacement. This record provides evidence of the durability of the RCP seals to withstand off-normal operating events (loss of seal cooling).

There were two other events at Palo Verde which establish the capability of the RCP seals to withstand loss of seal cooling events. These events are described in the CEOG report CEN-408 which was prepared in response to Draft Regulatory Guide DG1008 and are as follows:

Event Date: July 6, 1988
Plant: Palo Verde, Unit No. 1
Seal Type: CE/KSB

Description: Component cooling water and seal injection water were intermittently lost on RCP 2B for eight (8) hours on 7/6/88. The loss was caused by an auxiliary transformer loop transient. The seals reached 152°F after experiencing conductive heating through the pump shaft for approximately 6 hours. Seal failure did not result.

Event Date: March 3, 1989
Plant: Palo Verde, Unit No. 3
Seal Type: CE/KSB

Description: Unit 3 was at 100% power and was scheduled to come down for a refueling outage in the next few days. Due to a loss of site power all 4 RCP's experienced a loss of seal injection water and component cooling water (CCW), in addition the controlled bleed off (CBO) flow was inadvertently not isolated. These conditions lasted for approximately 90 minutes, seal temperatures reached 437°F.

Seal damage to pump 1B was evident by abnormal CBO/staging pressure after reestablishment of seal injection. Following reestablishment of seal injection water two of the RCP's were started to establisn forced circulation in the RCS and run approximately 7 to 8 hours at RCS normal operating conditions

Response 440.120 (Cont'd):

with subsequent run time at decreasing RCS temperature and pressure for cooldown, which took approximately 29 hours. External seal leakage from pump 1B was later verified to be 1.25 GPM. Only pump 1B experienced leakage. Seals in all four pumps were replaced. Again, subsequent examination of the seals did not reveal any evidence of the seals "popping open" phenomenon mentioned in DG 1008.

(NOTE: This event is listed in NUREG-1401, Appendix A).

The above additional information provides further credence to the CESSAR-DC position that the System 80+ RCP shaft seals are highly reliable and are capable of limiting seal leakage to a maximum of 8 gpm per pump for at least 7-8 hours in the unlikely event that alternate AC power is not available and a station blackout occurs.

It should be noted, however, that the System 80+ primary design basis for coping with station blackout and other loss of sealing cooling events is to maintain seal injection water flow to the seals as described in our responses to questions 440.118 and 440.119.

Question 440.121:

Recently, Arizona Public Service (APS) personnel identified a potentially significant interfacing system LOCA (ISLOCA) event on the Palo Verde RCP seal cooling system while reviewing NRC Information Notice No. 89-54 "Potential Overpressurization of the Component Cooling Water System." By letter dated January 18, 1991, APS notified the NRC of the potential for a small break LOCA due to a tube rupture in the RCP high pressure seal cooler (HPSC). A HPSC tube rupture would be classified as an ISLOCA and results in overpressurization of the CCW system. The overpressurization of the CCW would result in a CCW surge tank relief valve release of RCS inventory onto the roof of the auxiliary building. The possibility of a HPSC tube rupture and its subsequent effects were not considered in the original design and is a previously unanalyzed event for the System 80 design.

Additionally, this event may impact GSI-23 for the System 80+ design since a postulated catastrophic HPSC tube rupture may simultaneously initiate degradation of RCP seals of the affected pump because cooling and lubrication flow would be diverted to the break. Therefore, the safety analysis for seal cooler tube rupture scenarios should include at least the following information:

(1) Evaluate for fuel damange under this case of small break LOCA conditions with:

  (a) Leak through only the ruptured seal cooler tube.

  (b) Leak through the ruptured seal cooler tube in conjunction with RCS leakage through a complete failure of the RCP seal stage assembly of the affected pump.

(2) The staff has evaluated the Palo Verde HPSC analysis (Ref. letter Trammell to APS issued 03/12/91) and has concluded that use of the leak-before-break (LBB) methodology is not applicable to a seal cooler tube rupture. The NRC LBB methodology is based on data of pipes 4 inch in diameter or larger. Due to uncertainties in the elastic-plastic fracture mechanics and the accuracy of the radiation monitoring system (RMS) for detection of small leaks under transient conditions (based on RMS experience of steam generators), the LBB methodology is not applicable for pipes that are less than 6-inch in diameter. Therefore, a non-mechanistic approach to seal cooler tube failure and consequence analysis should be used for the System 80+ design.

(3) Assess the radiological consequences and determine if the event is within a "small" fraction (10 percent) of the 10 CFR Part 100 guidelines. Use the apropriate criteria for such a failure as described in the Standard Review Plan (SRP) NUREG-0800 Section 15.6.2 "Radiological Consequences of Failure of a Small Line Carrying Primary Coolant Outside Containment." Evaluate if the assumptions made in CESSAR-DC Section 15.6.2 for input parameters and initial conditions are the most limiting conditions for a seal cooler tube rupture.

(4) Incorporate this scenario into the System 80+ PRA as appropriate.

Question 440.121 (Cont'd):

(5) Evaluate if the IRWST will have sufficient volume to permit operators to conduct an orderly RCS cooldown and depressurization under leak rates determined for item Nos. 1(a) and 1(b) of this RAI.

(6) Identify design features and associated emergency procedure guidelines for the System 80+ design that would prevent or mitigate the potential for overpressurization of the CCW system due to a seal cooler tube rupture.

(7) Propose any needed design modification to mitigate the consequences of a seal cooler tube rupture without terminating seal cooling/injection.

Response 440.121:

Recently, Arizona Public Service (APS) personnel identified a potentially significant interfacing system LOCA (ISLOCA) event on the reactor coolant pump (RCP) seal cooling system while reviewing NRC Information Notice No. 89-54 "Potential Overpressurization of the Component Cooling Water System." APS notified the NRC of the potential for a small break LOCA due to a tube rupture in the RCP high pressure seal cooler (HPSC). This HPSC tube rupture would be classified as an ISLOCA and would result in an overpressurization of the component cooling water system (CCWS) which eventually results in a CCWS surge tank relief valve release of reactor coolant system (RCS) inventory onto the roof of the auxiliary building. It was further stated that the possibility of a HPSC tube rupture and its subsequent effects were not considered in the original design and is a previously unanalyzed event for the System 80 design.

Based on the design criteria used for the HPSC as stated in our response to question 440.123, C-E believes that a HPSC tube rupture or the combination of a HPSC leak and a RCP seal failure is highly unlikely, however, System 80+ will be designed to accommodate the RCP HPSC event and potential overpressurization of the CCWS by incorporating the following design criteria:

1) the CCWS will be able to accept the maximum in-leakage expected from a RCP HPSC tube rupture without overpressurizing the CCWS by appropriately sizing the existing CCWS HPSC relief valve and

2) the CCWS HPSC relief valve discharge will be contained within containment to prevent significant release of radioactivity to the environment and therefore, within a small fraction of the 10 CFR Part 100 guidelines.

(1) The results of the Steam Generator Tube Rupture presented in section 15.6.3 of CESSAR-DC demonstrate that for RCS leaks up to 440 gpm, departure from nucleate boiling does not occur and all acceptance criteria are met. This flow rate bounds those expected for the HPSC leak and the combination of a HPSC leak and a RCP seal failure.

(2) It was stated that the leak-before-break (LBB) methodology is not applicable to a seal cooler tube rupture. C-E agrees that the LBB methodology will not be applied to the seal cooler tube rupture event.

Reponse 440.121 (Cont'd):

(3) The CCWS HPSC relief valve discharge will be contained within containment and, therfore, there will be no significant release of radioactivity to the environment. As a result, this event is within a small fraction of the 10 CFR Part 100 guidelines. The assumptions made in CESSAR-DC Section 15.6.2 for input parameters and initial conditions are the most limiting conditions for a Double-Ended Break of a Letdown Line Outside Containment.

(4) The HPSC tube rupture scenario is considered to be a small break LOCA and is already covered in the System 80+ small break LOCA PRA.

(5) The discharge of the CCWS HPSC relief valve will be directed to the Containment Floor Drain Sump which is within the Holdup Volume. The Holdup Volume has a spillway to the In-Containment Refueling Water Storage Tank (IRWST). When the Holdup Volume reaches 60,000 gallons, water spills over to the IRWST thereby replenishing the IRWST water volume. Therefore, no matter what the leak rate to the CCWS is during a RCP HPSC tube rupture, the operators will have sufficient water volume in the IRWST to conduct an orderly RCS cooldown and depressurization.

(6) Appropriate sizing of each RCP HPSC relief valve to accept the maximum expected in-leakage from a HPSC tube rupture will prevent overpressurization of the CCWS. The Emergency Procedure Guidelines for a Loss of Coolant Accident for System 80+ will be fundamentally similar to those provided in "Combustion Engineering Emergency Procedure Guidelines," CEN-152, Revision 03. These guidelines are adequate for this scenario. The leak can be detected by a radiation detector which taps off of the CCWS pump outlet or by a rising surge tank level. The CCWS surge tank has a high level alarm to alert the operators of an abnormal CCWS surge tank level. Furthermore, on the primary side of the HPSC there are temperature indicators and high temperature alarms on the inlet and outlet of the HPSC which will also be used to diagnose the event. The leak can be isolated by shutting the HPSC tube side isolation valves and/or by shutting the CCWS isolation valves for the affected RCP.

(7) The design will mitigate the consequences of a seal cooler tube rupture without terminating seal cooling/injection by (1) properly sizing the CCWS HPSC relief valve to accept the maximum expected in-leakage from a RCP HPSC tube rupture without overpressurizing the CCWS, and (2) directing the discharge from this relief valve to the Containment Floor Drain Sump.

Question 440.122:

Evaluate the probability and the radiological consequences associated with (1) a RCP seal failure resulting in a throttle seal cooler (TSC) tube rupture and (2) throttle seal cooler tube rupture independent of a seal failure. Use the guidance of the previous RAI (440.121) for the HPSC tube rupture scenario.

Response 440.122:

C-E has been asked to evaluate the probability and the radiological consequences associated with (1) a reactor coolant pump (RCP) seal failure resulting in a throttle seal cooler (TSC) tube rupture and (2) a TSC tube rupture independent of a seal failure. It should be noted that a seal failure will not cause a TSC tube rupture because the RCP TSCs are designed for reactor coolant system pressure. Also, C-E is asked to use the guidance of the previous RAI (440.121) for the HPSC tube rupture scenario.

System 80+ will be designed to accommodate a (RCP) throttle seal cooler (TSC) tube rupture and a potential component cooling water system (CCWS) overpressurization. The design will incorporate the following criteria:

1) the CCWS will be able to accept the maximum in-leakage expected from a RCP TSC tube rupture without overpressurizing the CCWS by appropriately sizing the existing CCWS HPSC relief valve (which protects the HPSC and the TSC CCWS from overpressurization), and

2) the CCWS HPSC relief valve discharge will be contained within containment to prevent a significant release of radioactivity to the environment and therefore, within a small fraction of the 10 CFR Part 100 guidelines.

The results of the Steam Generator Tube Rupture presented in section 15.6.3 of CESSAR-DC demonstrates that for RCS leaks up to 440 gpm, departure from nucleate boiling does not occur and all acceptance criteria are met. This flow rate bounds those expected for the TSC leak and the combination of a TSC leak and a RCP seal failure.

C-E will not apply the LBB methodology to the TSC tube rupture event.

The CCWS HPSC relief valve discharge will be contained within containment, and therefore, there will be no significant release of radioactivity to the environment. This event is within a small fraction of the 10 CFR 100 guidelines. The assumptions made in CESSAR-DC Section 15.6.2 for input parameters and initial conditions are the most limiting conditions for a Double-Ended Break of a Letdown Line Outside Containment.

The TSC tube rupture scenario is considered to be a small break LOCA and is already covered in the System 80+ small break LOCA PRA.

Response 440.122 (Cont'd):

The discharge of the CCWS HPSC relief valve (which protects both the HPSC and TSC CCWS side from overpressurization) will be directed to the Containment Floor Drain Sump which is within the Holdup Volume. The Holdup Volume has a spillway to the In-Containment Refueling Water Storage Tank (IRWST). When the Holdup Volume reaches 60,000 gallons, water spills over to the IRWST thereby replenishing the IRWST water volume. Therefore, no matter what the leak rate to the CCWS is during a RCP TSC tube rupture, the operators will have sufficient water volume in the IRWST to conduct an orderly RCS cooldown and depressurization.

Appropriate sizing of each RCP CCWS HPSC relief valve to accept the maximum expected in-leakage from a TSC tube rupture will prevent overpressurization of the CCWS. The Emergency Procedure Guidelines for a Loss of Coolant Accident for System 80+ will be fundamentally similar to those provided in "Combustion Engineering Emergency Procedure Guidelines," CEN-152, Revision 03. These guidelines are adequate for this scenario. The TSC leak can be detected by a radiation detector which taps off of the CCWS pump outlet or by a rising CCWS surge tank level. The CCWS surge tank has a high level alarm to alert the operators of a rising level.

The design will mitigate the consequences of a seal cooler tube rupture without terminating seal cooling/injection by (1) properly sizing the CCWS HPSC relief valve (which protects both the HPSC and the TSC CCWS side from overpressurization) to accept the maximum expected in-leakage from a RCP TSC tube rupture without overpressurizing the CCWS, and (2) directing the discharge from this relief valve to the Containment Floor Drain Sump.

Question 440.123:

Provide a structural/mechanical evaluation and the performance criteria for the design of the HPSCs and TSCs. In addition, include any discrepancies that may exist in the seal cooler criteria as compared to the EPRI URD Section 3.4.2.6.1 where the thermal barrier heat exchanger (System 80+ HPSCs and TSCs) should have a stress and fatigue analysis which addresses all anticipated steady-state and transient conditions, including pump in hot standby, pump start from hot standby, loss of restoration of seal injection flow, and pump operation with a degraded seal cartridge.

Response 440.123:

The high pressure seal cooler (HPSC) and throttle seal water (TSC) are designed and constructed in accordance with ASME Section III Subsection NB (Class 1) for the primary side and Subsection ND (Class 3) for the cooling water side of the HPSC and Subsection NB for the cooling water side of the TSC.

The design conditions are:

Primary Side:	Design Pressure	2500 psia
	Design Temperature	650°F
Secondary Side:*	Design Pressure	150 psig
	Design Temperature	200°F

* The design conditions for the secondary side of the throttle seal cooler (TSC) are 150 psig and 200°F, however, the pump supplier has elected to upgrade the secondary side of the TSC to 2500 psia and 650°F and construct it to Subsection NB because the TSC is attached directly to the pump seal housing which is designed for RCS conditions.

As part of the RCP design process a seal cooler design stress analysis is performed for both the HPSC and TSC in accordance with Paragraph NB 3400 for loads associated with design, normal, upset, faulted, test and transient conditions. The transient conditions include those the pump experiences from reactor coolant system transients plus loss of and restoration of seal injection water with pump operating and on hot standby and loss of and restoration of CCW with the pump operating and on hot standby. In addition an analysis is performed to demonstrate that the high pressure cooler and internal tube bundle is rigid and, therefore, not subject to cyclic fatigue due to vibration. A similar analysis is performed on the throttle seal cooler tube to demonstrate that it is rigid and also not subject to cyclic fatigue.

The design criteria used for the HPSC's and TSC's is consistent with that suggested in EPRI URD Section 3.4.2.6.1 with the following clarification. The thermal barrier heat exchanger is usually a heat exchanger mounted internal to the pump and in some cases integral to major pressure boundary components. As such the stress analysis and particularly the fatigue analysis is critical to pressure boundary integrity. In the case of the System 80+ RCP's, the thermal

Response 440.123 (Cont'd):

barrier function is performed by the high pressure seal cooler (HPSC) in combination with seal injection water as explained in the response to question 440.125. The HPSC is mounted external to the pump and is a more traditional shell and tube heat exchanger. The Palo Verde RCP HPSC and TSC were evaluated for cyclic loading and it was determined that the exclusion criteria of Paragraph NB-3222.4(d) was satisfied and a fatigue analysis was not required. The System 80+ HPSC design is the same as the Palo Verde HPSC.

Question 440.124:

(GSI-105: Interfacing Systems LOCA at LWRs)

The evaluation of the HPSCs and TSCs should consider the events described in 440.121 as potential interfacing system LOCA pathways. These heat exchangers should be included in the analysis for addressing ISLOCA under the guidance of RAI 440.47 and appropriately refl' 'ed in the resolutions of GSI-105 and GSI B-63.

Response 440.124:

C-E is asked to consider the reactor coolant pump (RCP) high pressure seal cooler (HPSC) and throttle seal cooler (TSC) tube ruptures as potential interfacing system LOCA pathways. Further, these heat exchangers should be included in the analysis for addressing ISLOCA under the guidance of RAI 440.47 and appropriately reflected in the resolutions of GSI-105 and GSI B-63.

The RCP HPSC tube rupture and the RCP TSC tube rupture will not be potential interfacing system LOCAs with a direct path release to the environment when the design changes as stated in 440.121 and 440.122 are incorporated into the System 80+ design. This position is based on the following:

1) there will be no significant release of primary coolant outside of containment via the component cooling water system (CCWS) because the RCP HPSC relief valve discharge is contained within containment. The HPSC relief valve protects both the HPSC and TSC CCWS sides from overpressurization;

2) the CCWS will not be overpressurized by this event due to appropriate sizing of the HPSC relief valve; and

3) any HPSC or TSC tube rupture that occurs can be isolated (see response to RAI 440.121 and 440.122).

Further, there will be no significant loss of reactor coolant system makeup water due to this event because the RCP HPSC relief valve discharge is directed to the Containment Floor Drain Sump which is within the Holdup Volume and which spills over to the In-Containment Refueling Water Storage Tank.

The resolutions of GSI-105 and GSI-63 are not affected by these design changes and do not have to be changed to accommodate these events. The CCWS is protected from overpressurization from a HPSC or a TSC tube rupture. Therefore, upgrading of the CCWS system piping and CCWS isolation valves to RCS design pressure are not required.

It is stated that the heat exchangers should be included in the analysis for addressing ISLOCA under the guidance of RAI 440.47, however, RAI 440.47 does not deal with ISLOCAs. RAI 440.47 deals with testing of the Shutdown Cooling System at full flow conditions. The RCP seal coolers are adequately designed for the seal cooler tube rupture event. The tube side of the HPSC and the TSCs are designed for RCS pressure while the shell side of the HPSC is designed for CCWS pressure and the shell side of the TSCs is designed for RCS

Response 440.124 (Cont'd):

pressure. If a rupture were to occur, the CCWS side (shell side) of the seal coolers would not be overpressurized because the seal cooler relief valve will be appropriately sized to prevent overpressurization for this event.

Question 440.125:

As part of the response to the above RAIs related to RCP seal integrity (GSI-23) and HPSC/TSC tube rupture analysis, provide a color coded simplified P&ID(s) of the CE-KSB RCP shaft seal system to be used in the System 80+ design clearly identifying:

(1) Each seal, each seal injection cooler (HPSC and TSCs), associated instrumentation and alarms, seal cooler heat exchanger isolation valves.

(2) Seal injection flow directions throughout the shaft seal assembly, including the journal bearing, during normal seal injection and during loss of seal injection (with and without HPSC tube rupture), indication of points of seal injections, RCS controlled leakage, etc...

(3) All CCW interfaces with the RCP seal injection and associated piping, components, and CCW instrumentation that would prevent or mitigate the radiological consequences of a catastrophic tube break in the seal coolers (HPSC and TSCs).

Response 440.125:

The System 80+ RCP P&ID is shown in Figure 5.1.2-2 of CESSAR DC. The flow paths, instrumentation and components (HPSC, TSC's and HPSC isolation valves) are schematically shown. The pressure and temperature entering each seal cavity is indicated and alarmed in the control room. The temperature entering and leaving the HPSC is also indicated and alarmed. The seal controlled bleed off flow is indicated and alarmed in the control room. All of these parameters can be recorded for the purpose of trending seal performance. Operating limits for these parameters are established and plant operators can evaluate seal condition and performance instantaneously or on a long term basis.

The System 80+ RCP uses a system of three seals, Figure 1, to seal the main shaft. The seals are supplied with filtered, chemically controlled seal injection water by the Chemical Volume Control System (CVCS). Two hydrodynamic seals are mounted in series, with a third stage vapor seal mounted above the two lower stages. Reactor coolant system (RCS) pressure is reduced to volume control tank pressure in stages by the controlled leakage bypass system, which contains throttling devices which are also called throttle seal coolers (TSC). The first two seals provide approximately 84 percent of the system pressure drop (42 percent each). The pressure drop across the third seal is approximately 16 percent. Each of the three seals is capable of operating at full system pressure.

Controlled bypass leakage is approximately 4 gpm and is piped to the volume control tank. Any leakage past the vapor seal is piped t the reactor drain tank. Seal materials consist of carbon for the rotating r ng and tungsten carbide for the stationary ring.

Response 440.125 (Cont'd):

The RCP seals are normally furnished with both CCW and seal injection (SI) water to provide independent and redundant seal cooling. The RCP seals are capable of continuous operation with either CCW or SI water and, therefore, loss of either system will not compromise the integrity of the seals.

RCP seal cooling is accomplished by incorporating a coolant recirculation system within the RCP. SI water is injected directly into the recirculation system. The system contains the HPSC which provides redundant cooling for the recirculated water by means of CCW on the secondary side of the HPSC. The seal coolant recirculation system is a feed and bleed arrangement, i.e., 6.6 gpm of SI water is feed in and 4.0 gpm is bleed out through the seals as controlled bypass leakage and the remaining 2.6 gpm passes into the RCP casing and then into the RCS. Total recirculation within the system is approximately 25 gpm.

Operation of the seal cooling recirculation system is described below.

Flow diagrams for normal operation of the seal cooling recirculation system with SI & CCW, with loss of SI water and with loss of CCW are shown on Figures 2 and 3. For normal operation, Figure 2, seal injection water (6.6 gpm) enters into the high pressure piping, mixes with the recirculated coolant and is directed to the HPSC before entering into the seal system. For this condition the primary source for cooling is SI water and the heat load on the HPSC is low. The TSC's are located before the second and third seal stages and provide supplementary cooling for these stages.

The recirculation water (25 gpm) from the HPSC enters the high pressure side of the first seal and is divided into two flow paths. A portion of the flow (21 gpm) is pumped through the journal bearing by the auxiliary impeller. This cools the journal bearing and minimizes the ingress of contaminants into the seal system. Approximately 2.6 gpm of this flow enters the RCS through the pump casing. The balance of the flow (18.4 gpm) recirculates back to the HPSC but mixes with 6.6 gpm of SI before the flow enters the HPSC. The second flow path (4.0 gpm) is through a TSC to the high pressure side of the second seal. Flow from the second seal continues through another TSC to the third seal and then to the volume control tank (VCT) in the CVCS. This controlled bypass leakage through the TSC's is commonly called controlled bleed off (CBO) flow (4.0 gpm).

In the event seal injection (SI) water is not available, Figure 3, water (4.0 gpm) comes from the RCS, mixes with the recirculation flow and passes through the HPSC. As before, a portion of the water from the HPSC is circulated by the auxiliary impeller through the bearing, mixes with water from the RCS and then back to the HPSC. The other portion of the flow becomes controlled bleed off (CBO) and passes through the seals to the VCT. For this condition the HPSC operates under maximum design heat load and cools mixed RCS water down to seal operating temperature.

For a concurrent loss of seal injection water and a HPSC tube rupture, water from the RCS would flow into the HPSC and through the ruptured tube. The controlled bleed off flow would decrease since the flow takes the path of least resistance and would bypass the seals and pass into the HPSC.

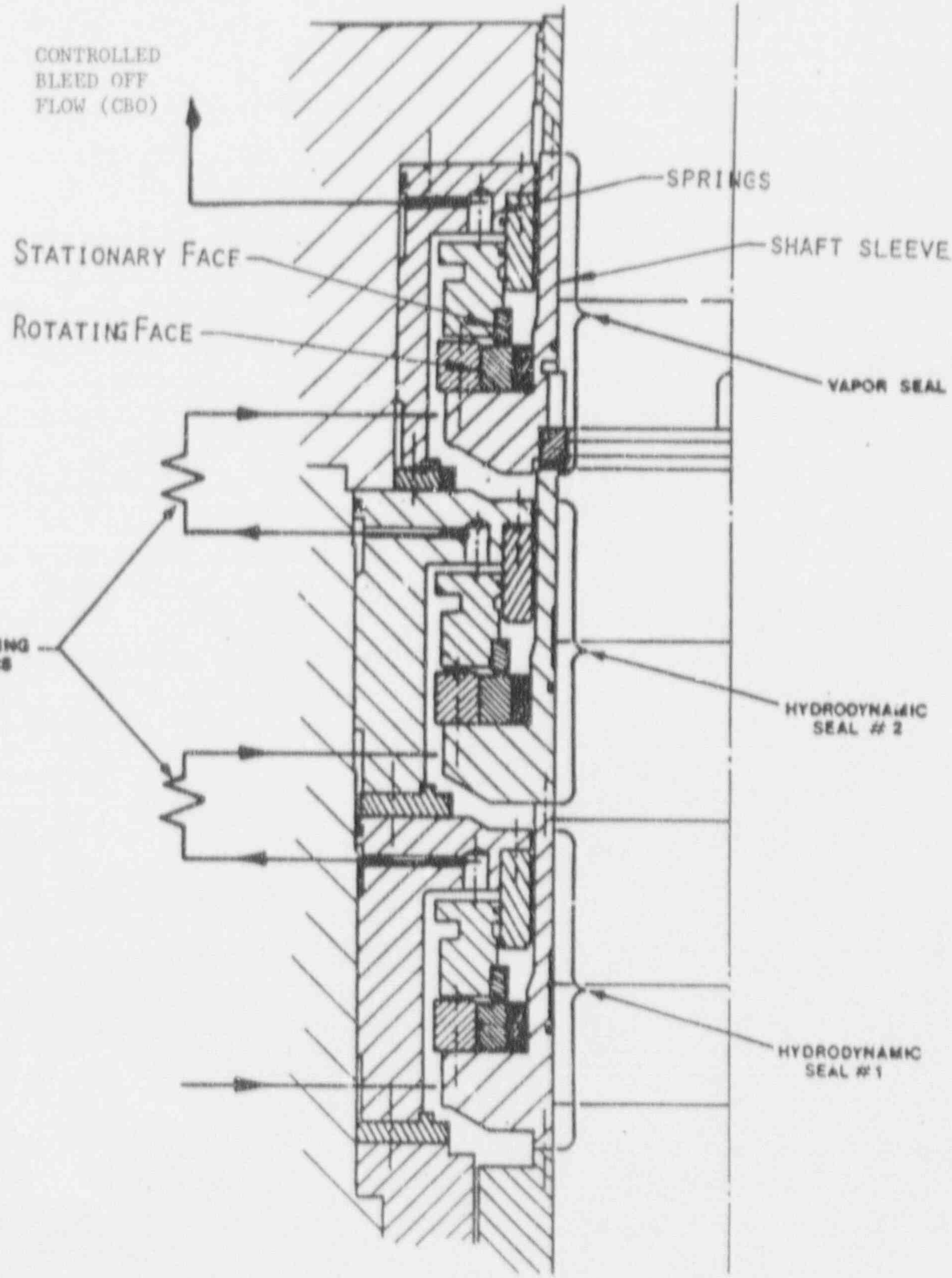
Response 440.125 (Cont'd):

If component cooling water is not available the seal cooling recirculation system operates the same as when SI and CCW is available except that the SI water provides all of the seal cooling. The flow diagram for this condition is the same as Figure 2.

High pressure isolation valves are provided upstream and down stream of the HPSC. These valves are designed to close against full system pressure. The mechanical integrity of the HPSC, the HPSC isolation valves and the piping connecting these components to the pump is assured by classifying them as Safety Class 1, ASME B&PV Code Class 1 components. The balance of the seal cooling recirculation system is contained within the pump pressure boundary seal housing which is a ASME B&PV Code Class 1 component.

All CCW interfaces with the RCP coolers (HPSC & TSC) are shown in Figure 9.2.2.1 CESSAR DC. A leak into the CCW system due to a seal cooler tube rupture can be detected by radiation detectors within the CCW system or by a rising CCW system surge tank level.

RAI 440.125



HYDRODYNAMIC SHAFT SEAL ASSEMBLY
FIGURE 1

RAI 440.125

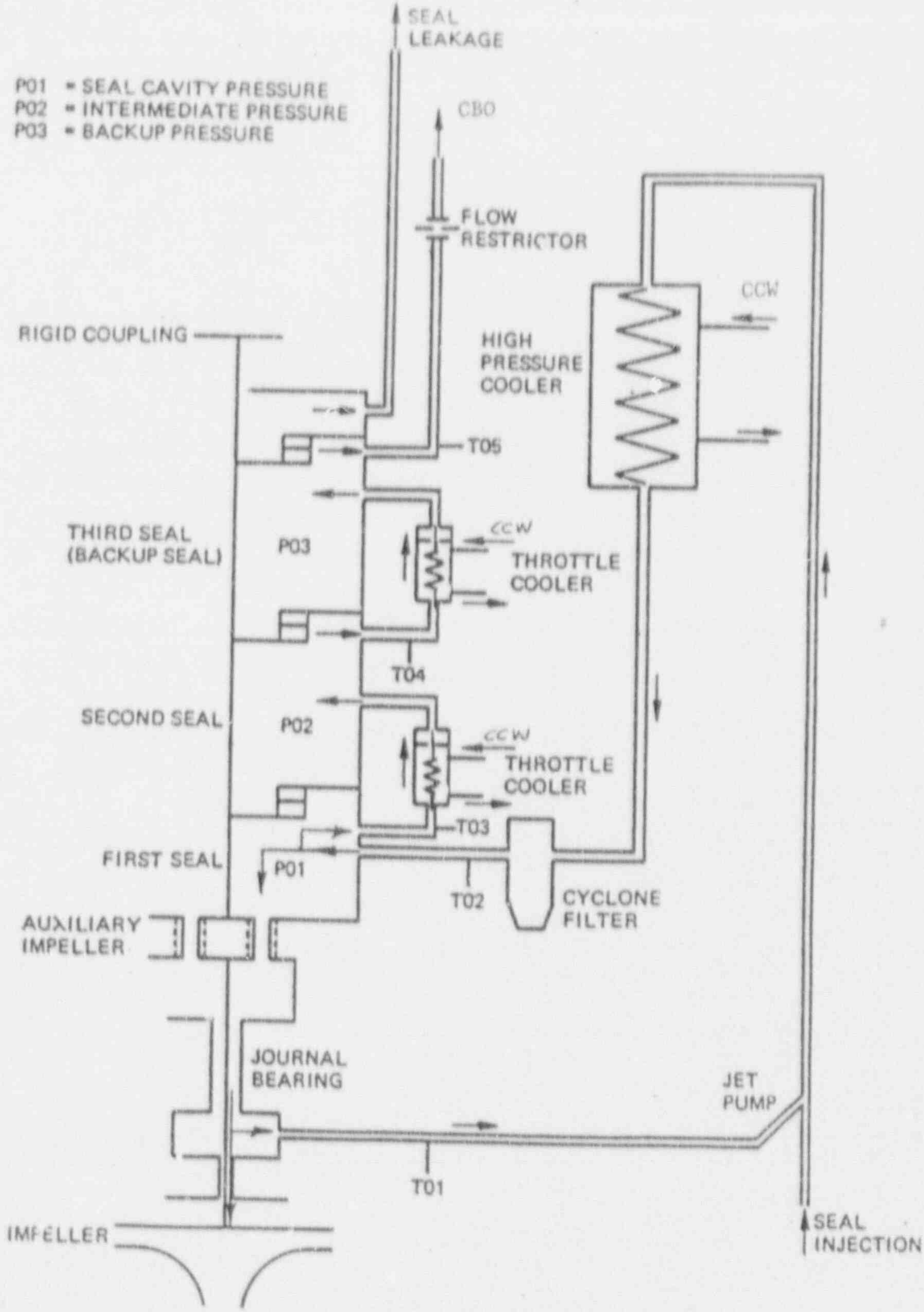


Figure 2 Flow Diagram for Hydrodynamic Shaft Seal System.
Normal Operation - with CCW & SI

RAI 440.125

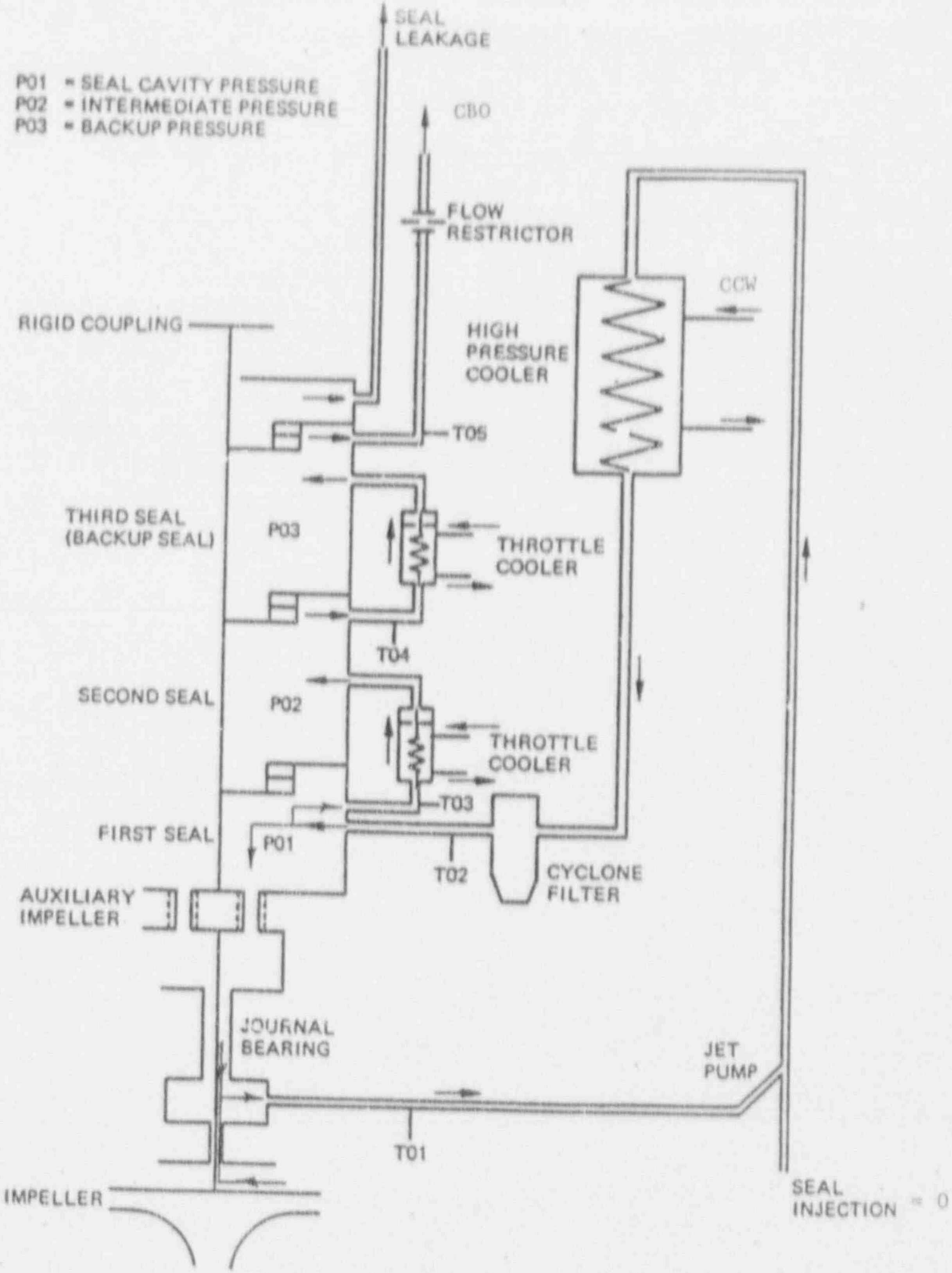


Figure 3 Flow Diagram for Hydrodynamic Shaft Seal System.
Operation without SI & with CCW

Question 440.126

Under LOCA conditions, if a loss-of-offsite-power (LOOP) occurred at some time interval after the emergency diesel generators (EDGs) are up to rated voltage and speed and after the required engineered safety features (ESF) actuations, the potential exists for draining the fluid systems during the time it takes to resequence, reload and restart pumps. Restarting essential pumps (safety injection and support systems) with voided lines may result in problems due to air/steam binding, pump over speed on low discharge resistance, or water hammer. Essential systems should be capable of successful restart following loss of offsite power, if a LOOP were to occur at the time of turbine trip or at anytime following the accident. Please evaluate the possibility and potential consequences of restoration of interrupted safety injection system (SIS) flow to the core with the following considerations:

1. Possibility of air/steam entrainment in the direct vessel injection (DVI) lines resulting in:

   a) Air/steam binding of the SIS pumps

   b) Water hammer on DVI lines and components after SIS pump restart due to steam voided lines.

2. Possibility of steam entrainment in the DVI line resulting in SI pump overspeed on pump restart after being sequenced onto the EDG bus/load resulting in an overload of the emergency diesel generators.

3. Possibility of draining other essential fluid systems such as the SW, CCW, Shutdown Cooling (SCS), Emergency Feedwater system, demonstrating that adequate safety system performance will be assumed for a delayed LOOP event.

Response 440-126

1a) The physical layout of the System 80+ plant (with the safeguards pumps directly below the IRWST) provides pump protection against air and steam entrainment upon loss of normal power. Equipment locations ensure a continuous positive pressure on the pump from both the suction and discharge side. The specific advantage of having the positive pressure is that there will be a water column seal protecting the pump. Backflow is prevented by a series of four check valves in the discharge header. The result is that, if power is lost, steam and air will be isolated from the pump, thus preventing binding.

1b) The question regarding water hammer in the DVI lines of the SIS is interpreted to be a result of the condensation-induced phenomena experienced in the feedlines and feedrings of steam generators. The scenario presented in the question conforms to the situation described in NUREG-1606. The issue identified in NUREG-1606 has been considered to be technically resolved with the incorporation of the design guidelines provided in NUREG-0918. These design guidelines have been used in the design of the DVI lines and, hence, the potential for condensation-induced water hammer is expected to be negligible.

The following is a discussion on how the DVI line design has incorporated the guidelines of NUREG-0918. These items will protect not only the DVI lines from condensation induced water hammer, but will also minimize steam and air induction into the SIS for pump protection.

The first recommended guideline in NUREG-0918 is to keep the piping flooded at all times so as to eliminate steam entrance into the system. Although this is an ideal situation for the injection line, it is not possible for all postulated LOCA scenarios. However, the design of the SIS provides two features that will minimize the effect of voids in the injection line.

The first is the time limit required on the control system to switch power sources and reestablish steady state flow. This will be discussed in more detail under the second design guideline. The intent is to minimize the amount of RCS inventory that would be lost during a power source transfer. The second is the physical interface differences between the SIS and feedring and their respective sinks. The feedring interface identified in NUREG-1606 discharges down through multiple holes located all along the ring. These holes act as orifices restricting flow out of the ring as the fluid level in the steam generator drops. The result is that if the fluid level drops at a sufficient rate, there would still be fluid in the ring above that in the generator leading to the phenomena detailed in figure 4 of NUREG 0918. The SIS interface with the RCS is not like the feedring interface with the steam generators. Instead, it is an abrupt pipe to vessel entrance without restrictions. Therefore, there is no orifice type interface and the fluid level will drop at the same rate in both the DVI line and the reactor vessel eliminating the fluid interface to seal the steam into the line upon refill.

The second guideline presented in NUREG-0918 is to minimize the time that the flowrate has been interrupted during power source transfer. This is where there is a major difference between the SIS and the situation presented in NUREG-1606. NUREG-1606 and NUREG-0918 discussion is based on a time frame of about 20 minutes before the restoration of steady state flow has been established. This allows for a significant drop in water level and, hence, steam entrainment into the feedring. However, the SIS design has invoked very stringent time requirements to restore full delivery flow subsequent to an interruption of power. Section 7.3.1.1.2.3.f-g, page 7.3-16 of CESSAR-DC details the requirements for establishing and reestablishing SIS flow subsequent to a loss in power. The time limit for System 80+ to establish flow at the onset of an ESFAS without offsite power is 40 seconds (worst case), and to transfer the SIS pumps to the diesel generators subsequent to the generators operating at rated speed and voltage is 20 seconds. These time frames are clearly within the guidelines of NUREG 0918.

The third design guideline presented in NUREG 0918 is to shorten the horizontal length of pipe connecting to the vessel. The intent is to minimize the potential volume in the injection line that can be filled with steam by providing a positive water seal. Horizontal lengths of piping are postulated to be susceptible to trapping steam resulting in water hammer.

The volume in the safety injection lines near the reactor vessel is minimized, as in this guideline, and also a reason related to the safety analysis. The safety analysis establishes limiting volumes with associated boron concentration levels for the Safety Injection System. Therefore, comparing the volume established in the safety analysis for the region closest to the reactor vessel to the volume of the feedring as shown in figure 3 of NUREG 0918, shows that the SI arrangement has limited the possible volume for steam entrainment. Furthermore, part of the volume credited for the safety analysis is beyond the first check valve. Hence the actual volume available for steam entrainment will be even less than that employed in the safety analysis.

2) The SI pumps are equipped with constant speed motors. Therefore, based on this and the protection against steam entrance described above in response to part 1, the SI pumps will not cause an overloading of the diesel generators due to overspeed following a LOOP.

3) The following is provided to demonstrate that the safety function of the essential systems listed in the question will not be jeopardized subsequent to a delayed LOOP event.

(a) CCW (Component Cooling Water System) - The CCW is a closed loop system with no direct interface with the primary system. All NSSS interfaces are across physical boundaries, e.g., heat exchangers, such that with a loss of power no fluid loss would occur. Therefore, the CCW system would not be drained and would retain its safety function subsequent to a delayed LOOP event.

(b) SCS (Shutdown Cooling System) - The SCS utilizes the same discharge interfaces as the SIS and takes suction from the bottom of the RCS hot leg line. Therefore, based on the description provided above to part 1 of this question the SCS system would accomplish its safety function subsequent to a delayed LOOP event.

(c) SW (Station Service Water System)- The SW is a closed loop system with no direct interface with the primary system. All NSSS interface is across a physical boundary, i.e. heat exchanger, such that with a loss of power no fluid loss would occur. Therefore, the SW system would not be drained and would retain its safety function subsequent to a delayed LOOP event.

(d) EFW (Emergency Feedwater System)- The EFW interfaces with the Feedwater system. As a result, the same guidelines of NUREG-0918 are applicable. The only notable difference is that the Emergency Feedwater pumps have been included in the same classification for the time limit to restore full flow as the SI pump of 20 seconds. Therefore, based on the description provided above to part 1 of this question, the EFW would not be drained and would retain its safety function subsequent to a delayed LOOP event.

QUESTION 440.127

(USI A-17: System Interaction in Nuclear Power Plants) USI A-17 deals with adverse systems interactions (ASIs) in nuclear power plants where intersystem dependencies (or system interactions) have been divided into three classes based on the way they propagate; functionally coupled, spatially coupled, and induced human intervention coupled ASIs as defined by NUREG-1299 and Generic Letter 89-18. USI A-17 is concerned with more than just water intrusion and internal flooding from internal sources since there are other coupling mechanisms, such as seismic events and pipe ruptures, that should be considered during the design phase ASIs review.

CESSAR-DC issue description of USI A-17 states that in NUREG-1229 the NRC concluded that for future plants, the existing SRP (NUREG-0800) in general covered the ASIs of concern. It should be noted that the NUREG-1229 conclusions were formulated without the benefit of a design specific review of an advanced design. New or differently configured systems (i.e., SDS, IRWST, SIS, non-safety CVCS, etc...) may not have an SRP section or have an SRP section requiring modification and subsequently lend the possibility for undiscovered ASIs. Therefore, please address the following item:

(1) Identification of provisions to be proposed in the System 80+ Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) program that account for identification and corrective actions for identified ASIs during the construction phase of a System 80+ plant.

(2) Propose a comprehensive and specific program providing:

(a) ASIs identification via the CESSAR-DC Appendix B PRA study.

(b) Full direction for implementing the resolution actions.

(c) Location of the program and findings to be incorporated in CESSAR-DC.

RESPONSE 440.127

(1) USI A-17 is responded to in CESSAR-DC Appendix A and,as a conclusion, states "the design process for the System 80+ Standard Design takes into account the possibility for interaction between systems to occur that may degrade plant safety, but are not easily recognizable. To the extent practicable, as the design progresses these interactions will be identified. Their impact on safety will be evaluated, and the necessary corrective actions will be taken."

RAI 440.127

The design process for System 80+ addresses the requirements to evaluate potential adverse systems interactions. A basic design requirement for plant general arrangements, safeguard systems and instrumentation was to maintain separation of components and power supplies so that adverse systems interactions, such as those identified in this RAI, would not occur. No adverse systems interactions have been identified nor are any expected.

As part of the normal design process, evaluation of the potential for ASI will continue. Any ASIs that are identified will be resolved so that the final design will not retain any ASIs which can have a significant impact on performance.

ITACC will be available to provide assurance that the facility is constructed and can be operated in conformity with the certified design. The ITACCs will be performed in conjunction with the tests and inspections required under the provisions of 10CFR Part 50. The scope of these combined test and inspection programs is such that ASIs not identified and resolved during the design process would be found in the course of executing the programs to bring the plant to an operational state. The impact of ASIs identified in this manner would be evaluated and corrective actions taken, as appropriate.

2(a) The System 80+ PRA directly covers functionally coupled ASIs. As part of a scheduled update of the System 80+ PRA, fire and flood risk potential are being qualitatively assessed. This will in part, address spatially coupled ASIs Spatially coupled ASIs are addressed in part by the seismic PRA. Induced human intervention coupled ASIs will be evaluated in parallel with the System 80+ PRA update.

2(b) Significant ASIs identified during the design process will be evaluated for their impact on plant safety. Appropriate design changes will be made to eliminate significant ASIs.

2(c) The program summary (part 1, above) will be added to CESSAR-DC Appendix A, USI A-17.

Direction for reviewing plant actual construction and operational data to evaluate the potential for ASIs will be incorporated in the detailed construction turnover and test procedures developed for the specific plant. Evaluation would be performed as part of the performance of system walkdowns, component and system operational testing, integrated system testing during steady state and transient testing and, finally, during test results review by plant technical teams.

ACCEPTANCE CRITERIA

The acceptance criterion for the resolution of USI A-17 is that attention shall be paid in the detailed plant design to detecting and minimizing the potential for ASIs due to the effects of flooding and water intrusion from internal plant sources, such as the incidents at operating plants referenced in NUREG-1174. The objective is to preserve the means for reaching and maintaining a safe hot shutdown.

In addition, consideration should be given using the overall plant PRA to identify ASIs, especially with regard to concerns based on operating experience documented in LERs and/or NRC Information Notices.

RESOLUTION

ASIs are difficult to predict or detect, and are determined by the specific, detailed system designs and layouts. They may also be influenced by building design features. I

For the System 80+ Standard Design, therefore, consideration is given during the development of the plant design to identifying and ameliorating potential ASIs, particularly with regard to flooding and water intrusion events which are not covered by current SRPs, as discussed in NUREG-1174. These events include water or moisture release from sources internal to plant structures, and may involve only small amounts of water and subtle communication paths to sensitive equipment such as electrical cabinets.

At the same time, the design is evaluated for its vulnerability to ASI's identified from previous designs or experienced at operating plants and reported in LERs and/or NRC Information Notices. This evaluation has been made for each of the interaction incidents resulting from water intrusion at operating plants described in the NRC Information Notices referenced in NUREG-1174, to identify the features of the System 80+ Standard Design which should ensure prevention of a similar interaction.

The analytical models developed for the System 80+ Standard Design PRA (CESSAR-DC Appendix B) have the capability to evaluate the impact of any systems interaction detected which appears to be significant. As the detailed design is developed, these analytical models will be used to identify potential ASIs and provide guidance on their elimination.

Insert A

Insert A

The design process for System 80+ addresses the requirements to evaluate potential adverse systems interactions. A basic design requirement for plant general arrangements, safeguard systems and instrumentation was to maintain separation of components and power supplies so that adverse systems interactions, such as those identified in this RAI, would not occur. No adverse systems interactions have been identified nor are any expected.

As part of the normal design process, evaluation of the potential for ASI will continue. Any ASIs that are identified will be resolved so that the final design will not retain any ASIs which can have a significant impact on performance.

ITACC will be available to provide assurance that the facility is constructed and can be operated in conformity with the certified design. The ITACCs will be performed in conjunction with the tests and inspections required under the provisions of 10CFR Part 50. The scope of these combined test and inspection programs is such that ASIs not identified and resolved during the design process would be found in the course of executing the programs to bring the plant to an operational state. The impact of ASIs identified in this manner would be evaluated and corrective actions taken, as appropriate.

QUESTION 440.134

Since the safety depressurization system is a safety grade system, propose technical specifications for SDS operability for all appropriate modes of plant operation (include full power conditions).

RESPONSE 440.134

The safety depressurization system technical specifications will be provided in a future amendment to CESSAR-DC, Chapter 16. See the response to the request for additional information from the NRC Technical Specification Branch, dated 10-16-91.

Question 440.136

(GSI-22: Inadvertent Boron Dilution Event)
In the event of a SCS or DVI component is taken out of service for maintenance and the line downstream of the respective component is drained, are there any SCS or DVI line interfaces that have the potential for inadvertently refilling these lines with deborated water?

Response

Inadvertent refilling of the DVI portions of the Safety Injection System (SIS) and the Shutdown Cooling System (SCS) with deborated water has been minimized to the extent practicable. The design of the SIS and the SCS has addressed the inadvertent boron dilution event presented in Generic Safety Issue 22 by controlling the system's source of water and preventing fluid back-flow into the system.

Specifically, all SIS and SCS fluid interfaces that supply makeup originate from a source of borated water of sufficient concentration to meet the system requirements. The IRWST is the source of safety grade water for all operations of the SIS and SCS. In addition, the SCS can be filled from the CVCS.

In addition, relief valve, vent and drain discharges are routed directly to one of three tanks (the reactor drain tank, equipment drain tank or the holdup volume tank) or to room sumps. The tanks are non-pressurized and are at a lower elevation than the pipe connection with the systems. This arrangement precludes the possibility of siphoning the contents out of a tank, or sump, and diluting the boron concentration in either system.

QUESTION 440.138

Explain any provisions of the System 80+ administrative procedures and Technical Specifications that justify the assumption that the 180 gpm flow of one CVCS charging pumps is a conservative dilution rate for MODE 5 midloop operations and not both charging pumps. In addition, since the non-safety related CVCS does not have any corresponding Technical Specifications for its operability or technically specified lockout provisions for a charging line not in use, provide justification for using one charging pump for operational modes 1, 2, 3, and 4 in the inadvertent deboration (ID) analysis.

RESPONSE 440.138

Powering of only one charging pump at a time is a design feature of the System 80+ CVCS design. In addition, administrative procedures will provide operator instruction on the proper operation of the standby pump. The maximum flow rate from one charging pump of 180 gpm is therefore justified for the Inadvertent Boron Dilution (IBD) analysis.