Log No: JCO-91-01-00

Justification
for
Continued Operation

PVNGS Unit 1

Potential for Small Break Loss of Coolant Accident

Due to Pipe Rupture in the Reactor Coolant Pump Seal Cooler

Revision 0

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Introduction

This Justification for Continued Operation (JCO) is applicable to PVNGS Unit 1 only, and is a revision to the JCO (Revision 1) provided to NRC in APS letter 161-03873, dated April 13, 1991. (Change bars are utilized throughout this JCO to identify the changes to the April 13, 1991, revision.) This JCO addresses the probability of a potential failure (and the consequences) of the High Pressure Seal Cooler (HPSC) piping during the remaining 4.5 month period of PVNGS Unit 1 operation, prior to the Unit 1 refueling outage, scheduled for February 1992. The differences between this JCO and the JCO (Revision 1) submitted on April 13, 1991, are:

- This JCO includes a human error rate for failure to restore power to the 13.8 KV non-class bus NAN-S01 (S02) in the event of a fast bus transfer failure.

- The event probability for Unit 1 is calculated separately from Units 2 and 3.

- The compensatory action administrative limit for the RCS equilibrium I-131 DOSE EQUIVALENT is 0.4 uCi/gm. This compensatory action limits the off-site dose of a catastrophic failure of the HPSC to less than 25% of 10 CFR 100 limits for the Generated Iodine Spike case.

Executive Summary

During review of NRC Information Notice No. 89-54 "Potential Overpressurization of the Component Cooling Water System," Arizona Public Service (APS) identified a scenario in which a break in the reactor coolant pump high pressure seal coolers (HPSC) could potentially result in a reactor coolant system (RCS) leak being released outside of the containment building. The scenario involves a leak from the reactor coolant pump HPSC into the lower pressure nuclear cooling water (NC) system. The resulting leak could potentially overpressurize the NC system. If this were to occur, and the NC containment isolation valves were unable to shut against the pressure or flow, and the operators were unable to identify the leaking seal cooler and isolate the leak with the seal cooler isolation valves, it could result in reactor coolant being discharged from the NC surge tank relief valve on the auxiliary building roof. APS has performed an analysis of this scenario and determined that continued operation is justified on the basis of the following:

There has never been a high pressure seal cooler leak in an ABB/Combustion Engineering plant. A search of INPO Nuclear Plant Reliability Data System (NPRDS), Nuclear Power Experience (NPE), and Licensee Event Reports (LER) data bases revealed only one instance of an inservice pressurized water reactor seal cooler pipe leak. That leak occurred in 1970 at the Beznau Unit 1, a Westinghouse plant in Switzerland.

APS performed a structural evaluation of the RCP seal cooler heat exchanger piping. The results of this evaluation are: (1) the stresses in the RCS seal cooler piping (ASME Class I piping) are low enough that no mechanism (for example vibration, thermal cycling, etc.) could be identified which would result in propagation of an existing flaw, and (2) the austenitic stainless steel used in the seal cooler piping is resistant to corrosion and erosion/corrosion damage.

The double ended shear of HPSC piping was evaluated to address concerns over possible fuel failure by examining the spectrum of break sizes evaluated for a small break loss of coolant accident (LOCA). The potential for additional leakage due to RCP seal degradation caused by the HPSC rupture was also included in the analysis. These analyses concluded that the combined leakage from a HPSC rupture and failure of all three stages of a RCP seal was bounded by the small break LOCA analyses and would not result in fuel failure. Additionally, it was concluded that there was sufficient Refueling Water Tank (RWT) volume to cooldown, depressurize, and isolate the leakage resulting from both a HPSC rupture and RCP seal failure.

A probabilistic risk assessment (PRA) of the worst case scenario was performed, without consideration for leak detection prior to failure. The PRA results indicate that the probability of having this event scenario at Palo Verde Unit 1 is 1.7E-5 per reactor year, or 6.4E-6 for the remaining 4.5 months of Unit 1 operation prior to the next refueling outage, scheduled to begin in February 1992. This event is considered a low probability event for non core melt sequences. For this event, the Standard Review Plan criteria for Exclusion Area Boundary (EAB) dose is a "small fraction"

(10%) of 10 CFR 100 limits. However, ANSI/ANS-51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, would classify this event, based on probability of occurrence, as a Plant Condition 5 event. The EAB dose criteria for a Plant Condition 5 event is the 10 CFR 100 limit of 300 Rem (Thyroid). The PVNGS Unit 1 administrative limit for RCS equilibrium I-131 DOSE EQUIVALENT (DEQ) of 0.4 uCi/gm results in an EAB dose of 61.3 Rem (Thyroid), which is less than 25% of 10 CFR 100.

As compensatory measures, appropriate operating procedures have been changed to require initiation of a plant shutdown within 4 hours of detecting RCS activity in the NC system. There are two methods available for detecting activity. The first method is an on-line radiation monitor which provides continuous monitoring of the NC system. This monitor will alarm in the control room within one hour (with current RCS activity levels) for a 0.1 gpm leak. The second method is a once per shift sampling of the NC system for activity. Radiochemical analysis of the sample provides confirmation of the radiation monitor operation and is capable of detecting a 0.1 gpm leak. If contamination of the system from short half-life fission products (indicative of an RCS leak) is detected, actions will be initiated during and following plant shutdown to locate and isolate the leak using the seal cooler isolation valves. The leak detection capability will ensure there is adequate time for operator action prior to the leak propagating to a point where the NC system would be overpressurized.

In addition, if the specific activity of the primary coolant is found to be greater than 0.4 uCi/gm I-131 DEQ for more than 48 consecutive hours, an orderly plant shutdown will be commenced and the unit will be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

In summary the above analyses and actions demonstrate that: (1) the probability of a leak occurring in the HPSC is very small, (2) if in the unlikely event a HPSC leak did occur it would be detected quickly and the unit would commence a shutdown and cooldown within 4 hours, (3) operations personnel can be expected to determine the location of a leak and have the ability to isolate the affected cooler effectively stopping the leak, (4) even if a catastrophic failure were to occur, with the compensatory action of prohibiting continued operation with equilibrium RCS activity levels above the administrative limit of 0.4 uCi/gm I-131 DEQ, and an iodine spiking factor of 500, the radiological consequence would be less than 25% of 10 CFR 100 limits for the Generated Iodine Spiking case. Thus, continued operation during the next 4.5 months will not adversely affect the health and safety of the public.

I. Equipment Description

A. Shaft Seal System

A mechanical seal arrangement is used to seal the reactor coolant pump shaft. This seal arrangement acts as a pressure boundary between the RCS and the containment while minimizing RCS leakage along the pump shaft.

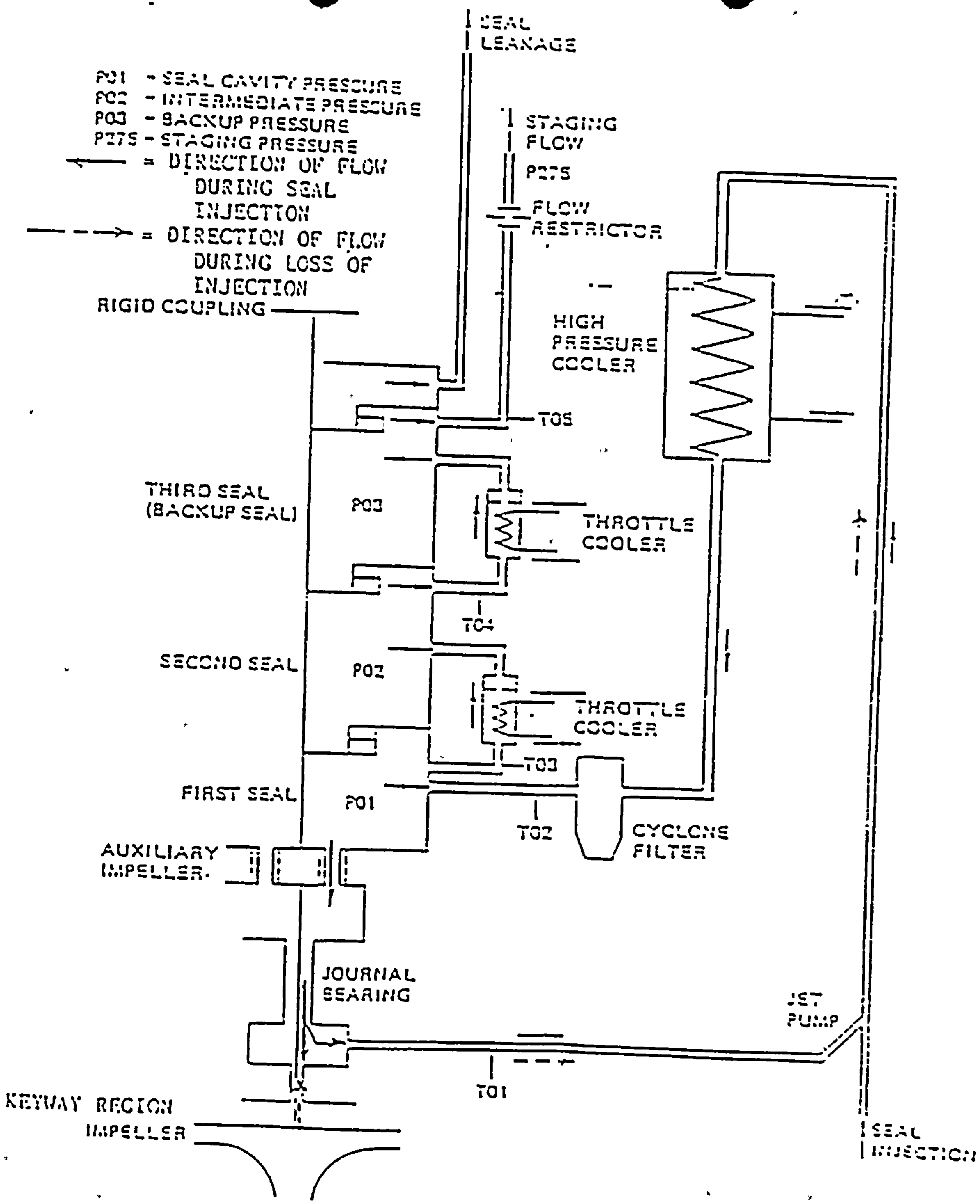
Mechanical seal operation depends on two basic elements. These are a rotating seal face which is attached to the pump shaft and a stationary seal face attached to the pump body. A liquid film is established between these two faces. This film layer acts to lubricate the faces thus preventing wear and also minimizes or prevents leakage. The seal materials consist of carbon for the rotating ring and titanium carbide for the stationary ring.

The CE-KSB pump uses a system of three seals (Figure 1) in series to accomplish sealing. Each seal by itself, is capable of providing full sealing capabilities. During normal operation however the pressure breakdown across the seals is divided. Each of the two hydrodynamic seals provide for a 42% pressure drop (945 psid) with the vapor seal providing 16% (360 psid). The pressure drop across the seals is established by the use of throttling devices in a controlled leakage bypass system. Controlled bleedoff leakage is normally 4 gpm and is piped to the Volume Control Tank (VCT). This controlled bleedoff is referred to as staging flow.

High pressure, filtered seal water is normally provided to the seals from the charging pumps. Seal injection water enters at a point below the water lubricated journal bearing, passes through a jet pump, and is directed to the high pressure cooler. The injection water acts as the driving fluid in the jet pump with the effluent from the water lubricated journal bearing being the driven fluid. This promotes circulation through the high pressure cooler and aids in cooling the journal bearing.

The effluent from the high pressure cooler enters the high pressure side of the first seal and is divided into two flow paths. The majority of the flow is pumped through the journal bearing by the auxiliary impeller, thus providing cooling for the bearing. A portion of this water then leaks past the shaft into the RCS. This prevents the ingress of contaminants from the RCS to the sealing system. The second flow path, referred to as staging flow, provides pressure staging and seal cooling. The staging flow path around the seals is required for seal pressure distribution. The staging flow continues through the two throttle coolers to the high pressure side of the third seal and then to the volume control tank.

Approximately 7 gpm of seal injection water at 120 °F is normally provided. Of this, 4 gpm is for controlled leakage with the remainder flowing into the RCS as mentioned above.



Flow Diagram for Hydrodynamic Shaft Seal System.

Figure 1

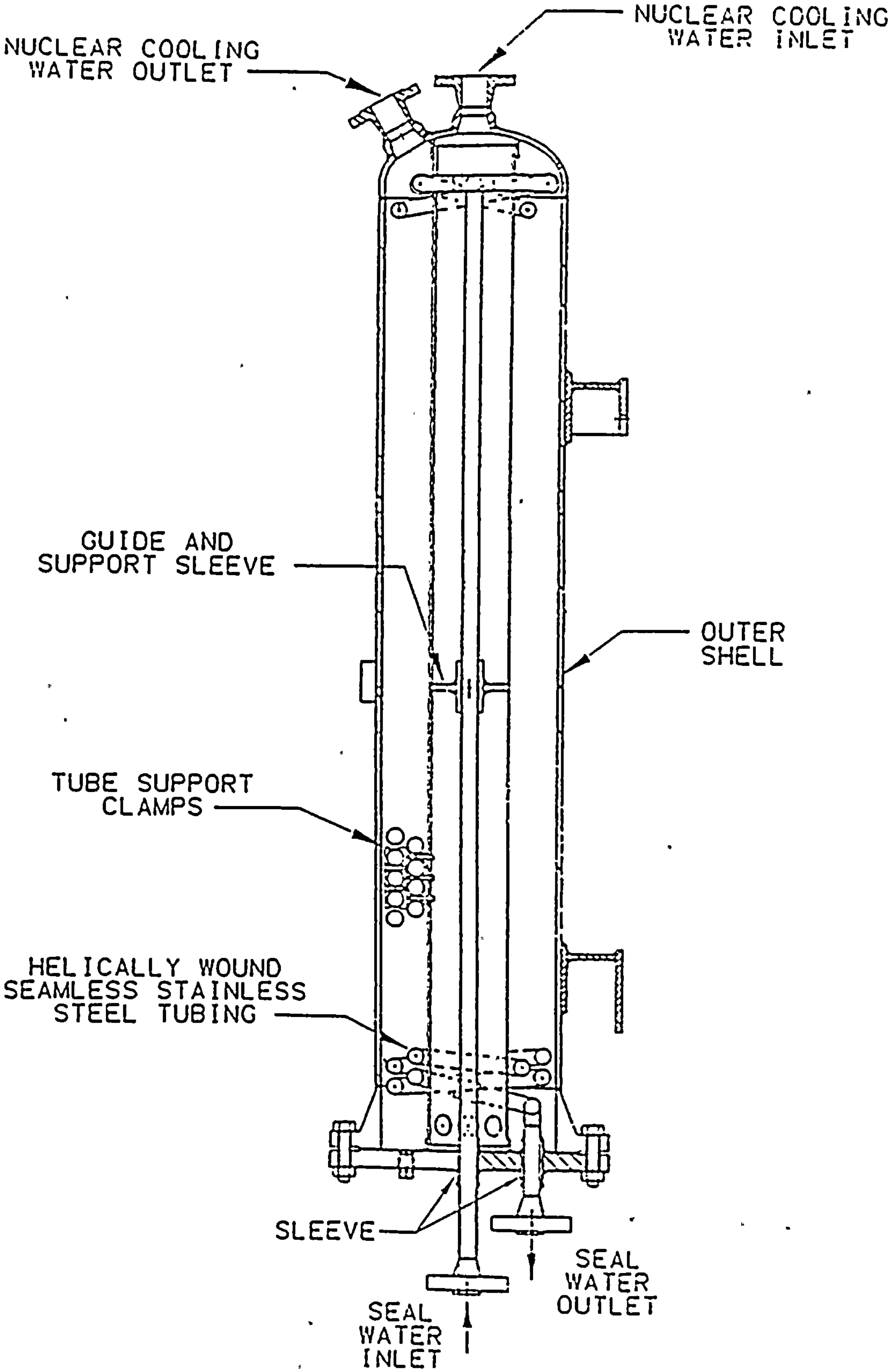
B. High Pressure Cooler

The High Pressure Seal Cooler (Figure 2) is composed of a 75 inch long, 12 inch diameter, outer shell (NC pressure side) with a seamless stainless steel, continuous, helically wound, 1.25 inch outside diameter Schedule 80 internal pipe (RCS pressure side). The high pressure piping enters the bottom through a sleeve which is seal welded on both the inside and outside to the pipe. The sleeve acts as a guide and brace for the high pressure pipe. The pipe continues straight up to the upper section of the shell passing through another sleeve which guides and supports it. At the upper section of the shell the high pressure pipe divides into two pipes helically wound downward through supports running the entire length of the cooler, and exits the shell at the bottom through a sleeve where the sleeve is again welded to the pipe both inside and outside the shell thus acting as a guide and support. The supports and clamps prevent movement of the high pressure piping.

Nuclear Cooling Water is introduced into the cooler at the top and channeled to the bottom of the shell. The nuclear cooling water channel has a number of diffuser holes at the bottom which redirect the flow upward and past the high pressure helically wound piping. The nuclear cooling water exits the shell at the top.

The cooler is built in accordance with ASME Section III, 1974 Ed. The primary (high pressure) coil is Class 1. The secondary side (shell) is Class 3. The internal pipe support/coil clamps are designed and constructed in accordance with Subsection NF for component supports.

The HPSCs are provided with inlet and outlet isolation valves on the RCS side. The isolation valves are 1 inch, motor operated, globe valves. The materials of construction are in accordance with ASME Section III requirements for Class 1 valves with a design pressure and temperature rating of 2,485 psi at 650°F. Operability and function are not impaired by SSE seismic loads. Isolation of the seal coolers to prevent exceeding RCS leakage limits was not considered an accident mitigation safety function in the original design basis, therefore, the motor operator and power supply are not Class 1E qualified. The motor operators are sized to close against a 2,500 psi differential pressure.



HIGH PRESSURE SEAL COOLER

Figure 2

C. Nuclear Cooling Water System

The NC system circulates water in a closed loop to collect heat from the normal operating nuclear-related equipment and reject it to the plant cooling (PW) water system. Two pumps, two heat exchangers, one surge tank, one chemical addition tank, and necessary piping, instrumentation, and controls comprise the components in the system.

The surge tank is located on the roof of the auxiliary building and establishes the net positive suction head pressure of the circuit. It serves as a reservoir for expansion and contraction of the cooling water and as a convenient location to introduce makeup to compensate for any system losses. Makeup is provided by the demineralized water system and the makeup line is sized for a nominal flow rate of 50 gpm. The surge tank is designed for a maximum pressure of 15 psig and has a relief valve with a setpoint of 10 psig.

Chemicals for corrosion and pH control are added to the cooling water via the chemical addition tank. To detect the level of radioactivity in the cooling water, the NC system is equipped with a radiation monitoring system which is further described below.

The following components are cooled by the NC:

Reactor coolant pump seal throttle coolers

Reactor coolant pump HP coolers

Reactor coolant pump thrust bearing oil coolers

Reactor coolant pump motor air and oil coolers

Letdown heat exchanger

Gas stripper

Boric acid concentrator

Radwaste evaporator package

Waste gas compressors and aftercoolers

Nuclear sampling coolers

Fuel pool cooling heat exchangers

Control element drive mechanism (CEDM) normal air cooling units (ACUs)

Normal chillers

Auxiliary steam vent condenser

Auxiliary steam radiation monitoring system cooling coil

Steam generator blowdown (nonnuclear) sampling coolers

The NC system provides cooling water to the RCP throttle coolers, cooling the seal water to the second and third stage seals. The seal water flows through 0.14 inch diameter tubing helically wound through NC cooled fluid chambers integral to the assembly housing. Postulation of a guillotine rupture of the throttle cooler tubing results in an estimated leak rate of only 2.1 lbm/sec. Leakage of this magnitude can be relieved through the NC system relief valves located inside containment without overpressurizing the system. Thus the NC containment isolation valves could be shut preventing any leakage from bypassing containment.

The components, other than the HPSC, which contain and/or interface with RCS fluid are located outside of containment and are isolable such that any consequences of a heat exchanger rupture would be within the existing design basis.

The nuclear cooling water system is a closed loop cooling system which provides cooling water to numerous heat exchangers that contain radioactive water. The NC system is constantly monitored by an on-line radiation monitor (RU-6) which alarms in the control room when the cooling water activity reaches a maximum preset level of 7E-6 uCi/gm. The radiation monitor is capable of detecting an RCS in-leakage of 0.08 gpm within one hour after the leak begins. If RU-6 is out of service chemical sampling will be performed which will detect this in-leakage rate within a period of 6 hours of leak initiation.

A qualitative assessment of the impact of the overpressurization of the NC system indicates that this postulated event does not challenge the integrity of the NC to essential cooling water (EW) isolation valves. The initial in-leakage into the NC system of 22 lbm/sec would displace the gas in the NC surge tank. The NC system has 38 relief valves with a cumulative relief capacity of approximately 480 gpm, which exceeds the in-leakage for the RCS seal cooler leak. Once the system goes solid the pressure will increase until the pressure reaches a steady-state condition equal to the setpoint of the highest set relief valve of 110 psig. Therefore the pressure rating of the system and the NC/EW isolation valves are not exceeded. The NC/EW isolation valves were hydrostatically tested with the disc in the fully closed position to 165 psig. The overpressurization will not extend into the safety-related EW system. If the surge tank were to fail, the steady state pressure in the NC system would be even less.

II. Safety Function

The only active safety function associated with the nuclear cooling water system is containment isolation, provided by the containment isolation valves and the connecting piping.

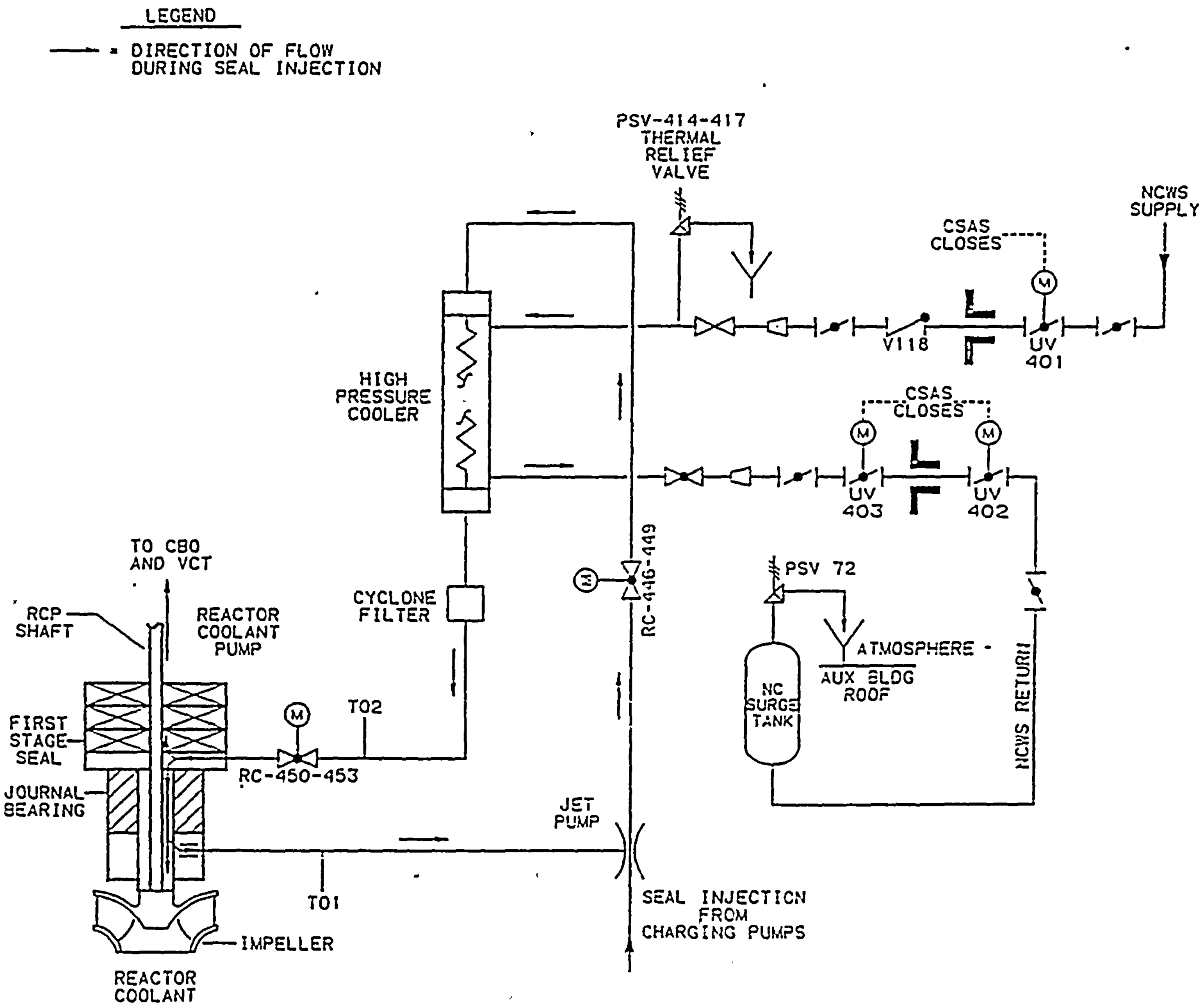
III. Postulated Event Scenario

The design basis of the HPSCs as described in the Combustion Engineering Standard Safety Analysis Report and the NRC Safety Evaluation Report was that any leakage from the RCS would be detected by a combination of the nuclear cooling water system radiation monitors and the high surge tank level switches. Both high radiation and surge tank level alarm in the control room. Once leakage is detected it would be isolated using the HPSC isolation valves. The possibility of a pipe rupture in the seal cooler and its subsequent effect on the nuclear cooling water system was not considered in the design.

During a review of NRC Information Notice 89-54 it was determined that a catastrophic failure of a high pressure seal cooler pipe could result in overpressurization of the NC system and the potential existed for a leakage path outside of containment. This failure would result in high pressure and temperature RCS fluid entering the low pressure, low temperature nuclear cooling water (NC) system piping. Figure 3 depicts a possible RCS leakage flow path outside containment via the NC system. Most of the RCS leakage would flow from the RCP body through clearances between the impeller hub and bearing sleeve, through a clearance between the bearing sleeve and stop seal, into a flow passage in the bearing sleeve, and through drilled clearances in the RCP seal housing. This leakage would then proceed to the HP Cooler via the HPSC inlet valve (and the associated jet pump). A parallel flow path would also be established past the journal bearing and the HPSC outlet valve. Calculations using two phase choked flow models, based on all the friction losses of the flow path, indicate the initial leakage flow rate through a doubled ended guillotine break of the pipe would be approximately 22 lbm/sec. Since containment isolation valves NC-401, 402 and 403 are not designed to isolate against RCS pressure, RCS fluid from the pipe failure is postulated to flow into the NC system providing a potential release path outside containment through the lowest set pressure relief valve (i.e., PSV-72) located on the NC surge tank on the auxiliary building roof. This relief valve (set at 10 PSIG) discharges to an open atmospheric scupper on the auxiliary building roof. Since the magnitude of the break exceeds the capacity of PSV-72, the design pressure of the surge tank (i.e. 15 psig) would be exceeded.

In addition to the above, a postulated catastrophic high pressure cooler pipe rupture may simultaneously initiate degradation of the RCP seals of the affected pump because cooling and lubricating flow would be diverted to the break. However, this degradation does not increase the radiological consequences of this event since any leakage would be confined to containment. Also, this flow would be available during the post LOCA recirculation phase of cooling, should it be needed.

This event would be a small break LOCA based on the criteria specified in procedure 4XEP-XZZ01, "Emergency Operations", and the operators would respond by entering and executing the actions from procedure 4XRO-XZZ08, "Small Loss of Coolant Accident." RCP alarm response procedures would direct the operators to close the seal cooler isolation valves to terminate the event. The valves are designed to operate against full RCS pressure, however they do not receive emergency power. If for some reason the affected HPSC could not be isolated, the RCS would be depressurized to allow the NC containment isolation valves to be closed isolating the leak.



HIGH PRESSURE COOLER
SEAL INJECTION AND NC FLOW PATHS

Figure 3

IV. Engineering Evaluations

A. Structural Evaluation of HPSC

The leak progression for piping is the development of a through wall crack which propagates in a circumferential direction. To initiate this progression there must be; (1) a flaw in the piping, (2) a forcing function to stress the pipe, (3) enough forcing function cycles to fatigue the pipe and cause the crack. Flaws may be the result of manufacturing defects, corrosion, or man made during installation or maintenance. Temperature changes, pressure changes, and vibration are examples of forcing functions.

1. High Pressure Seal Cooler Design and Manufacturing Evaluation - The seal coolers 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. The design conditions are:

Primary Side:	Design Pressure	2500 psia
	Design Temperature	650°F
Secondary Side:	Design Pressure	150psig
	Design Temperature	250°F

The seal cooler specification required design stress analysis in accordance with paragraph NB 3400 for loads associated with design, normal, upset, emergency, faulted and test conditions. In addition an analysis was performed demonstrating the cooler and internal pipe bundle to be rigid and therefore not subject to cyclic fatigue due to vibration.

The specification required the vendor to provide a manufacturing and testing records package to include certified mill test reports, radiographic evaluation reports, hydrostatic test results, non-destructive examination (NDE) results and ASME Manufacturers Data Reports. A listing of the contents of the records package and a Certificate of Equipment were reviewed by APS and the acceptance criteria were met. These ASME requirements for Class 1 components provide assurance as to the high integrity of the seal coolers.

Engineering has also reviewed the operational work history of the high pressure seal coolers. The results of the review indicated that no pipe leak problems to date have been reported for the seal coolers in any of the three units.

ASME Section III Subsection NB, paragraph NB-3222.4, requires the performance of a cyclic loading analysis on Class I components. Paragraph NB-3222.4 (d) allows exclusion of the fatigue analysis if six conditions are met. The cooler design successfully met the code exclusion criteria, and therefore no additional analysis was required. Engineering reviewed the calculations performed to support the exclusion, to determine the sensitivity of the calculation as it might apply to the development of a leak. The exclusion analysis included the design thermal and mechanical loading cycles imposed on the cooler during the life of the plant. A conservative number of plant start-ups and shutdowns, loss of seal injection, and loss of cooling water events were assumed in the analysis. A review of the type of loadings which might occur in the time period between detection of a leak and shutdown of the plant was performed. Using the same criteria used in the original analysis it was determined that sufficient design margin exists and no adverse cyclic loading is probable.

The high pressure cooler design report included a vibration analysis of the cooler assembly and of the pipe bundle. The cooler assembly was included in the overall model of the high pressure cooling water piping system. The report indicated a minimum frequency of 35.2 Hertz for the piping. This places the piping in the rigid frequency range.

A finite element vibration analysis of the cooler pipe bundle was also included in the HPSC design. The analysis resulted in a first fundamental frequency of 33.08 Hz, confirming the rigidity of the piping configuration. Since the design of the cooler minimizes the fatigue cycles it is very unlikely that any flawed piping inside the reactor coolant pump seal cooler would develop a leak or break.

2. Potential High Pressure Seal Cooler Degradation Mechanisms - Two potential damage mechanisms were evaluated for applicability to the seal coolers. Stress corrosion cracking and erosion/corrosion of Type 347 austenitic stainless steel piping were determined not to be damage mechanisms which would degrade the integrity of the coolers. Stress corrosion cracking (SCC) is a damage mechanism that requires the simultaneous action of a corrosive agent and a sustained tensile stress. SCC may occur in combination with hydrogen embrittlement, whereby the atomic hydrogen interacts with the metal to induce subcritical crack growth leading to fracture. Austenitic stainless steels are generally considered immune to hydrogen embrittlement. The HPSC piping material is manufactured to the ASME SA-213 TP 347 specification for seamless piping. The Type 347 material is a solution annealed, carbide stabilized, austenitic stainless steel which provides superior resistance to SCC and hydrogen embrittlement. Additionally, strict chemistry controls on the RCS are maintained to limit the presence of halogens and dissolved oxygen. Chemicals for corrosion control and pH are also added to the nuclear cooling water system via the NC chemical addition tank.

Erosion/corrosion is a flow assisted form of corrosion. This form of degradation occurs primarily in carbon steel applications as a result of the generation of a passive film (e.g.magnetite) on the internal surface of the pipe, which can be removed by the flowing water. Important parameters influencing this phenomena are material properties, hydrodynamics, and water chemistry. Austenitic stainless steels are considered very resistant to erosion/corrosion compared to carbon steel because of the their stable chromium oxide surface film. The fluid flow velocity through the seal cooler piping is only approximately 9 ft/sec. Primary water chemistry requirements include maintaining pH above 6.9 and dissolved oxygen below 0.1 ppm (steady state). Consideration of these parameters indicates that the seal cooler piping is not likely to be subject to erosion/corrosion.

3. Detailed Stress Evaluation - Maximum load conditions on the piping system, including the 1-1/4 inch diameter pipe, were evaluated and found acceptable by meeting Code limits in the original design calculations. The present review assures that this system, in particular at its weld to the blind flange, is not susceptible to excessive or unusual mechanical or thermal loads that could accelerate any cracking mechanisms that may be present. Pressure surges from the pump are minimal, as well as temperature transient excursions during operations. There are no quick acting valves in line with the pipe or component that could create a water or steam hammer effect in the piping system. These operational conditions define the cyclic range to begin at start-up, through normal plant operations and then "cold" shutdown. The number of plant operational cycles were chosen conservatively to be fifty cycles in order to obtain the worst case Usage Factor for any of the three PVNGS Units.

In addition to the original piping design calculations which demonstrated Code compliance, further stress evaluation of the pipe was performed at its weldment, in the heat affected zone, where it joins to the plate. Again the pipe was found to comply with the requirements of the B & PV Code, Article NB-3200 for fatigue evaluation. Design calculation, 13-MC-RC-515, demonstrates satisfaction of Code limits for both primary and local secondary stresses when the pipe is subjected to normal operational mechanical and thermal loads and for normal with safe shutdown earthquake conditions.

The calculation for fatigue life of the pipe for plant startup/shutdown shows that the HPSC pipe can withstand more than 1.5E6 of these cycles, with an extremely small Usage Factor of 2.8E-5. Here the summation of the alternating stress and the mean stress is greater than the yield stress, σ, but less than twice the yield stress, 2σ. Which means that the pipe juncture at the plate will "shake down" to elastic behavior after a few plant cycles and the true mean stress is calculated to be very much less than yield stress. The alternating peak stress is taken as half of the apparent nominal linear stress range but not greater than the yield stress on the inside surface of the pipe and much below yield on the surface directly opposite on the outside of the pipe. From these calculations it is demonstrated that the fatigue condition for maximum stresses can only occur in the elastic domain after shake down and any crack growth must occur in this elastic domain. Further, the primary (equilibrium) stresses are largely composed of thermal expansion stresses which under the rules of NB-3200 are local secondary stresses. So when the pipe is operational the mechanical stresses are small and in the absence of any sustained load on the pipe, throughout its cycling, it can not go plastic nor have any plastic increments or ratcheting. The entire pipe is not subject to any collapse condition.

Any crack growth can only be by unstable tearing as a direct function of the cyclic elastic fatigue mode. Since Palo Verde Unit 1, has had no more than thirty, (30), startup/shutdown cycles with no cracking or leakage in evidence, the potential for a catastrophic failure of the pipe is remote.

<u>B. Probabilistic Risk Assessment (PRA) of Catastrophic HPSC Pipe Rupture</u>

This scenario is evaluated from the standpoint of event likelihood and probability of failure to isolate the break. The PRA results reflect a probability of exceeding a "small fraction" of

10 CFR 100 limits if a HPSC tube rupture event were to occur during the next 4.5 months of PVNGS Unit 1 operation. The dose, however, will not exceed 25% of 10 CFR 100 limits.

1. Event Likelihood - The likelihood of a seal cooler pipe break is evaluated by review of both generic failure data and prior operating experiences based on the NPRDS and LER data bases. Generic failure rates for heat exchangers and small bore piping (< 3 inch) range between 5E-5 and 4E-4 per year. These generic failure rates utilize expert opinion, due to the lack of pipe rupture evidence. A value of 1E-4 per year is chosen as the best estimate frequency for a RCP seal cooler pipe break. This corresponds to one in ten thousand years. NPRDS search showed no prior US PWR RCP seal cooler pipe failure in approximately 25 million RCP Operating hours. There was one known seal cooler pipe leak event in Beznau Unit 1 (Switzerland). The RCP manufacturer was Westinghouse.

Pipe failure and degradation mechanisms are dependent on items such as material, age, chemistry and flow conditions. The seal coolers at Palo Verde are made of the austenitic stainless steel, which is resistant to intergranular stress corrosion cracking (IGSCC) and erosion/corrosion damage. These coolers have been in service since the mid-80s and no existing piping flaws from design, manufacturing, and installation which could potentially lead to a seal cooler pipe failure have been identified. Potential age degradation on pipe or welds should not be a consideration because of relatively short in-service hours of these coolers compared to some other PWR plants.

Even though uncertainties exist in the pipe break estimate because of no prior occurrence, differences in PWR seal cooler design and age, the 1E-4 per year per High Pressure Seal Cooler estimate should conservatively reflect our current state of knowledge on failure frequency.

2. Mitigation - The High Pressure Seal Cooler pipe leak or break is isolable via closure of both MOVs on each cooler. Motive power is supplied by non-class 1E power. If off-site power is available during the event and fast bus transfer is successful or the operator provides backup to the automatic fast bus transfer, the operators can close these valves and isolate the break. The combined probability for failing to diagnose the event and terminate the break flow is 4.3E-2. If power is available to the HPSC MOVs, the human error rate used for failing to diagnose the event and initiate mitigating actions in two hours is conservatively estimated as 1E-2. If power to the HPSC MOVs is lost due to fast bus transfer failure, a 2.54E-1 non-recovery factor was used.

It is possible that dual relief paths could exist through both the NC system and seal bleedoff due to postulated seal degradation. If this were to occur, the break flow and the total inventory lost from the refueling water tank to outside containment is small enough that it would not interfere with the establishment of containment sump recirculation, if needed. Therefore, seal degradation results in no increased risk from this particular scenario.

3. Total Calculated Frequency - The total frequency is calculated by considering the number of RCP High Pressure Coolers, the likelihood of having a pipe break, and the plant's capability to mitigate. The result is as follows:

For Unit 1: 1.7E-5 per year (about one in sixty thousand years).

During the next 4.5 months (anticipated time to Unit 1 refueling outage when the modification will be implemented) the probability of having this event scenario at Palo Verde Unit 1 is 6.4E-6. This is considered a very low probability for non core melt sequences, and the potential risk imposed on public health and safety during the JCO duration is, therefore, judged to be minimal.

V. Safety Evaluation

A. Transient Analysis of Inter-System LOCA (ISLOCA) Event

An analysis of the double ended shear of a HPSC pipe was performed by the Safety Analysis group. This evaluation was performed to address two concerns. Whether fuel failure would be expected to occur as a result of the small break LOCA created by the pipe failure, and whether the Refueling Water Tank (RWT) volume was sufficient to allow an orderly cooldown and depressurization.

The double ended shear of HPSC piping was evaluated to address concerns over possible fuel failure by examining the spectrum of break sizes evaluated for a small break loss of coolant accident (LOCA). A failure of the RCP HPSC would correspond to a break size of 0.000975 ft^2 resulting in an initial leak rate of approximately 230 gpm. Additional leakage may occur as a result of RCP pump seal degradation due to loss of cooling and lubrication as a result of the HPSC failure. This leakage to containment is expected to be less than 120 gpm with all three seal stages failed. Thus the total leakage from the RCS would be less than 350 gpm. The break sizes evaluated for the small break LOCA analyses were from 0.02 ft^2 to 0.5 ft^2. Break sizes in this range do not result in fuel failure. The 0.5 ft^2 break area (an approximate 56,150 gpm blowdown) bounds the expected leakage rate from both the HPSC rupture and RCP seal leakage due to loss of cooling. Based on this analysis, the high pressure seal cooler failure event would not result in fuel failure even if additional leakage from the RCP seals occurred.

1. Identification of Events and Causes - Direct release of reactor coolant may result from a break in the RCP high pressure seal cooler, as previously described. Leakage will continue as a function of the RCS pressure and cold leg inlet temperature until the RCS is cooled and essentially depressurized.

2. Sequence of Events and Systems Operation - Table 1-1 presents a chronological list of events which occur during the ISLOCA event, from the time of the failure of the RCP high pressure seal cooler until shutdown entry conditions are achieved.

This analysis assumes that the operator trips the unit immediately in response to RCP trouble alarms[1]. Thereafter, no further operator action is assumed for the first 30 minutes of the transient. This analysis also assumes a loss of off-site power at t=0 and a single failure of one diesel train at t=30 minutes. These failures maximize the leak rate throughout the transient and result in the availability of only one safety train during the cooldown portion of the transient.

At 30 minutes, the operator is assumed to begin a controlled cooldown of the RCS at a cooldown rate less than the administrative limit of 75°F/hr. The highest cooldown rate assumed in the analysis was 66°F/hr.

Operator actions (summarized in Table 1-1) to achieve shutdown cooling are based on maintaining pressurizer level control, and a subcooling margin[2] of at least 28°F, by manipulating high pressure safety injection (HPSI), charging and letdown, the pressurizer auxiliary spray, the atmospheric dump valves (ADVs), and the auxiliary feedwater pump.

3. Mathematical Model - A transient analysis of the ISLOCA event was performed using the CE Plant Analysis Code (CEPAC) to simulate the thermal-hydraulic response of the nuclear steam supply system (NSSS). CEPAC is a "best-estimate" code derived from the CESEC code described in Chapter 15 of the PVNGS UFSAR. It was selected for use in this analysis because:

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[1] Note that the radiological release calculation allows 600 seconds of additional leakage before the operator identifies the event and trips the reactor. To compare the CEPAC times and radiological release calculation scenario times add 600 seconds to the indicated CEPAC time.

[2] Administrative limit is 28°F during the controlled cooldown.

a. CEPAC could be used interactively to model operator actions during the cooldown portion of the event, from the initiation of a controlled cooldown at 30 minutes to completion of Mode 5 entry requirements. The CESEC code does not provide this flexibility.[3]

b. CEPAC could be used to model the ISLOCA event by adapting the input requirements for a small cold leg break event to conservatively represent the leak rate resulting from the ISLOCA, and by redefining selected edits to provide the parameters needed to calculate the ISLOCA dose (i.e., the leak rate and enthalpy as a function of time).

c. A PVNGS database for CEPAC was previously developed, which was used as a basis for evaluating the ISLOCA event.

Based on these considerations, the CEPAC code provides a timely and effective means of evaluating the ISLOCA event, and an acceptable basis for calculating the associated radiological dose.

4. Input Parameters and Initial Conditions - The assumptions and initial conditions for the ISLOCA event are summarized below.

Core Power Level, % rated power	100
RCS pressure, psia	2,400
Core inlet coolant temperature, °F	570
Core mass flow rate, % rated flow	95
Pressurizer liquid level (nominal)	53
Steam generator pressure, psia (nominal)	1,107
Equivalent Critical flow area, ft^2	0.000975

[3] The CESEC based cooldown algorithm previously used for long-term cooldown events was also evaluated for this application, but was deemed unsuitable because input changes needed to reflect the ISLOCA event would require extensive revision and recertification of the code.

These parameters were selected to maximize the primary system mass release from the ISLOCA. The equivalent critical flow area noted above was selected to match or exceed the calculated critical flow using Henry-Fauske's model over the RCS pressure range from normal operation to shutdown entry conditions. The calculated leak rate of 9.6 lbm/sec at an RCS pressure of 400 psia is assumed constant, for pressures below 400 psia, until the RCS is sufficiently cooled to preclude further leakage.

5. Results

The dynamic behavior of important NSSS parameters following an ISLOCA is presented in figures 4-1 through 4-8. This analysis assumes that the reactor coolant pumps (RCPs) are manually tripped off-line at 0.0 seconds. The initial leak rate of 22 lbm/sec (230 gpm) exceeds the capacity of the charging pumps. As a result, the pressurizer pressure gradually decreases from an initial value of 2,400 psia. The primary leak rate and drop in pressurizer water level cause the pressurizer heaters to come on. The third charging pump turns on, and the letdown flow drops to its minimum value. With all three charging pumps running, the pressurizer pressure and level continue to drop due to loss of inventory. This results in the pressurizer heaters turning on to maintain pressurizer pressure above 2,200 psia.

At 1800 seconds, the operator begins depressurization of the RCS to establish HPSI flow, and initiates a controlled cooldown using the ADVs, one auxiliary spray valve, a charging pump, a HPSI pump, and one auxiliary feedwater pump. Shutdown cooling entry conditions (400 psia, 350°F) are achieved at 4 hours 26 minutes after the reactor trip.

The postulated seal cooler pipe rupture may lead to degradation of the RCP seals as cooling and lubrication flow would be diverted to the break. If this were to occur, the total leakage of the primary water will include the leakage through the break and the leak flow through the degraded seal. In order to quantify this seal leakage, two different sources of information were consulted.

The first source, the station blackout analysis, assumed 100 gpm leakage for four pumps or 25 gpm per pump for the seal leakage. The second source, the RCP pump manual, presents a 120 gpm expected leakage for the failure of all three seals at a pump pressure of 2,250 psia. The largest of these two leak rates, 120 gpm, combined with the expected leakage from the HPSC rupture results in a total leak rate of 350 gpm. The UFSAR presents the largest break area for the small break LOCA analyses as 0.5 ft^2 in the pump discharge leg. The initial blowdown rate from a break this size is greater than 5,400 lbm/sec at 2,250 psia and 600°F average temperature, approximately 56,150 gpm. This blowdown rate is more than 150 times the expected leak rate for both failure of the HPSC and RCP seal. The small break LOCA analyses shows that for the 0.5 ft^2 break area, there is no cladding rupture and associated fuel failure. Therefore, no fuel failure would be expected even if there is seal degradation as a result of the seal cooler pipe rupture.

The ISLOCA run using CEPAC has shown that the 4 hour 26 minute integrated leakage is 222,000 lbm. At 4 hours 26 minutes the RCS temperature and pressure have decreased to below shutdown cooling entry conditions, less than 350°F and 400 psia. Assuming conservatively an additional four hours of cooldown after initiating shutdown cooling, the additional leakage flow to the atmosphere during this time period can be calculated. Note that the primary coolant will eventually stop leaking to the low pressure side of the seal cooler once the RCS pressure drops to atmospheric pressure. The leakage rate drops from an initial 22 lbm/sec to 9.6 lbm/sec at the end of 4 hours 26 minutes due to decreasing RCS pressure. The 9.6 lbm/sec leakage was conservatively assumed to continue for the next four hours, resulting in an additional leakage of 138,240 lbm. Thus a total of 360,240 lbm or approximately 43,200 gallons of RWT water will be required to maintain RCS inventory.

Technical specification 3/4.5.4 and Figure 3.1-2 of Technical specification 3.1.2-6, require a minimum of 573,744 gallons of water in the RWT. UFSAR, Table 6.2.1-6, presents that a minimum of 64,386 ft^3 (481,607 gallons) of water is needed for cooling in the event of a LOCA. Thus a margin of about 92,000 gallons exists between the required and available quantities of water. If a RCP seal leakage were to occur as a result of the HPSC leak, the water from the RCP seal would remain in containment and would be available for containment recirculation. The 43,200 gallons leaked to the atmosphere as a result of the HPSC failure would be the only water not available for recirculation. Since this is less than the 92,000 gallons of excess RWT capacity, adequate water is available for cooling in the event of both a HPSC rupture and consequent RCP seal failure.

The inventory demand on the Condensate Storage Tank (CST) for a natural circulation cooldown interval of 13.33 hours has been determined to be 168,000 gallons, in response to NRC Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," Rev. 1. This volume includes sufficient inventory for RCS sensible heat removal, steam generator inventory makeup, and decay heat removal until shutdown cooling entry conditions are established. The CST minimum required volume of 300,000 gallons is therefore adequate to meet this demand.

Table 1-1

Sequence of Events for an ISLOCA Due to
Failure of a RCP High Pressure Seal Cooler

Time (sec)	Event	Setpoint or Value
0	RCPs are tripped	--
4	Reactor trip occurs on low pump speed, % flow	95
	Loss of main feedwater pumps	--
288	Pressurizer backup heaters come on, begin cycling, psia	2,200
1,800	Operator begins depressurization, opens ADVs to maintain subcooling margin and establish HPSI flow	--
2,300	HPSI actuation signal, psia	1,837
2,320	HPSI flow begins	--
15,960	Cooldown rate < 75°F/hr Mode 5 Shutdown entry conditions established	--

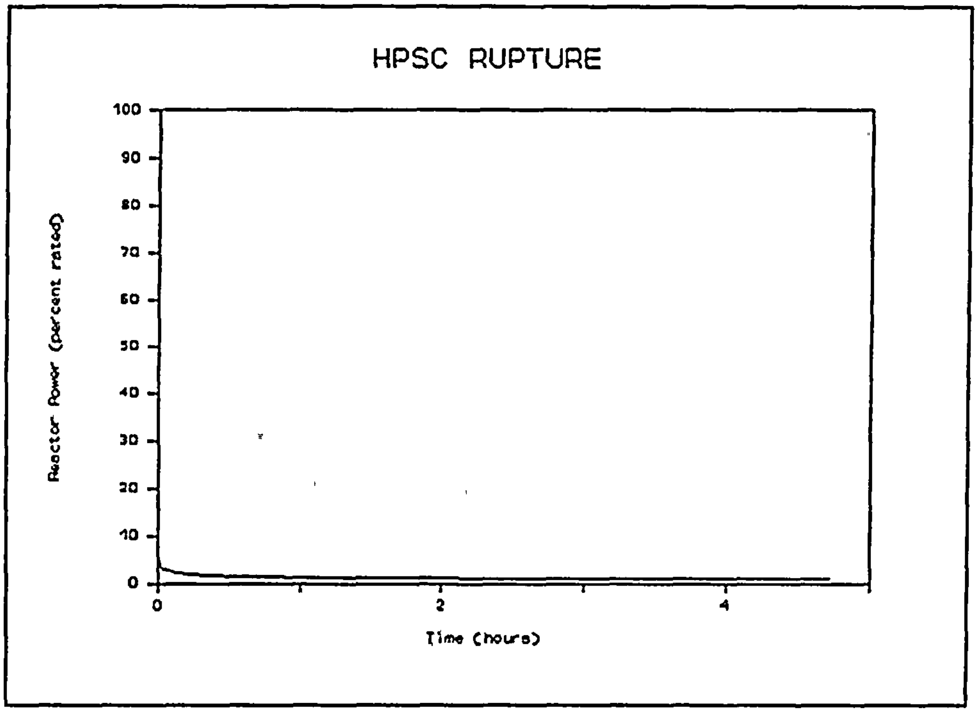

Figure 4-1 Reactor Power

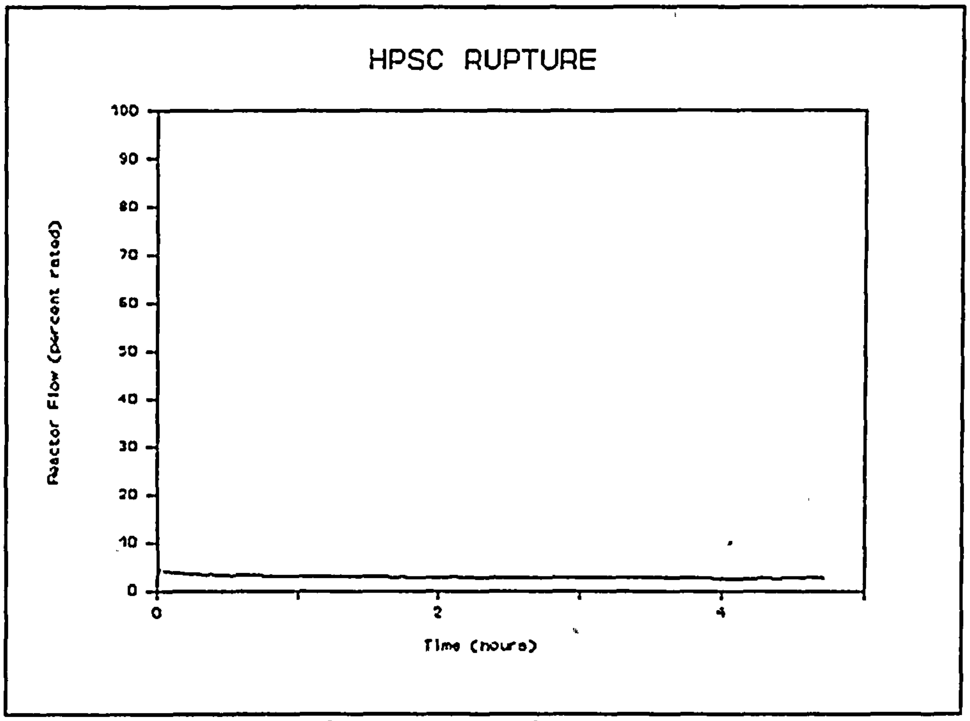

Figure .+-2 Reactor Coolant System Flow

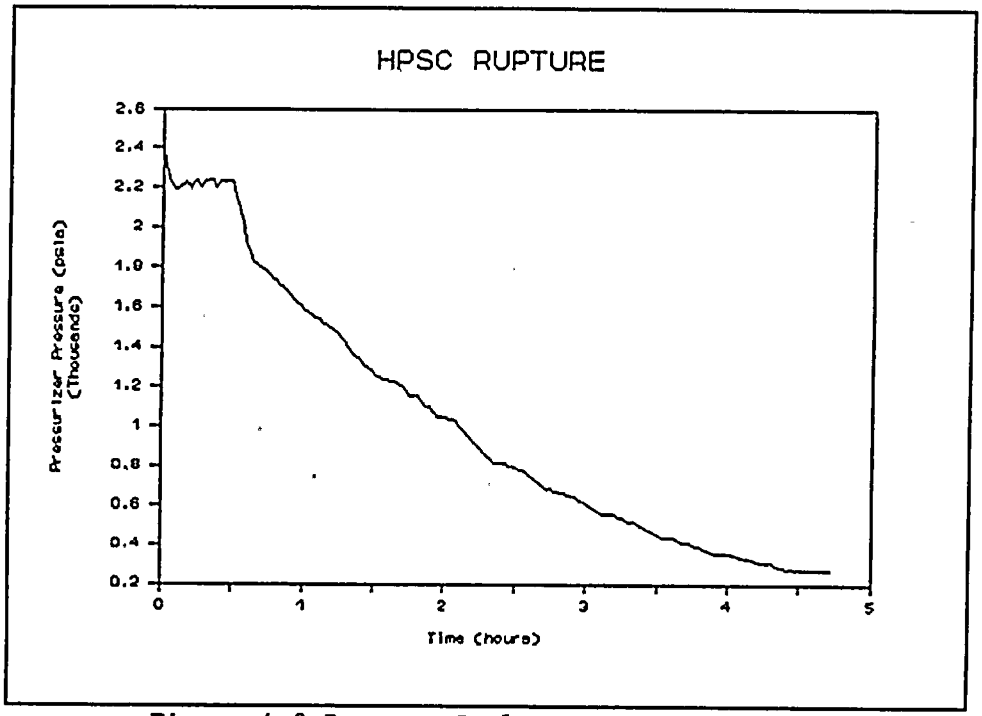

Figure 4-3 Reactor Coolant System Pressure

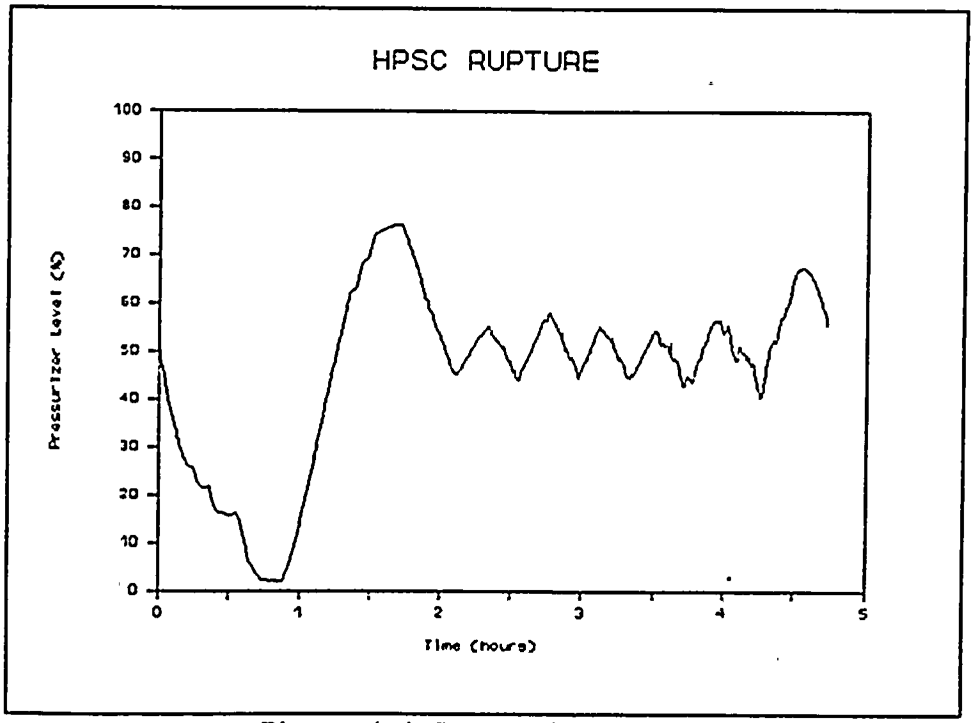

Figure 4-4 Pressurizer Level

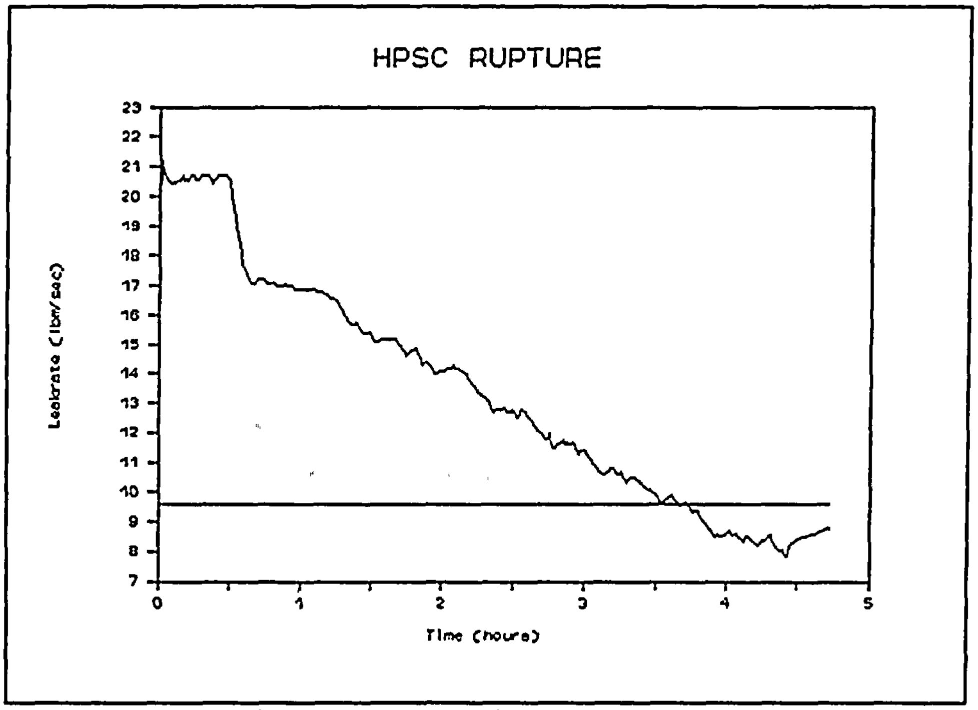

Figure 4-5 Reactor Coolant System Leak Rate

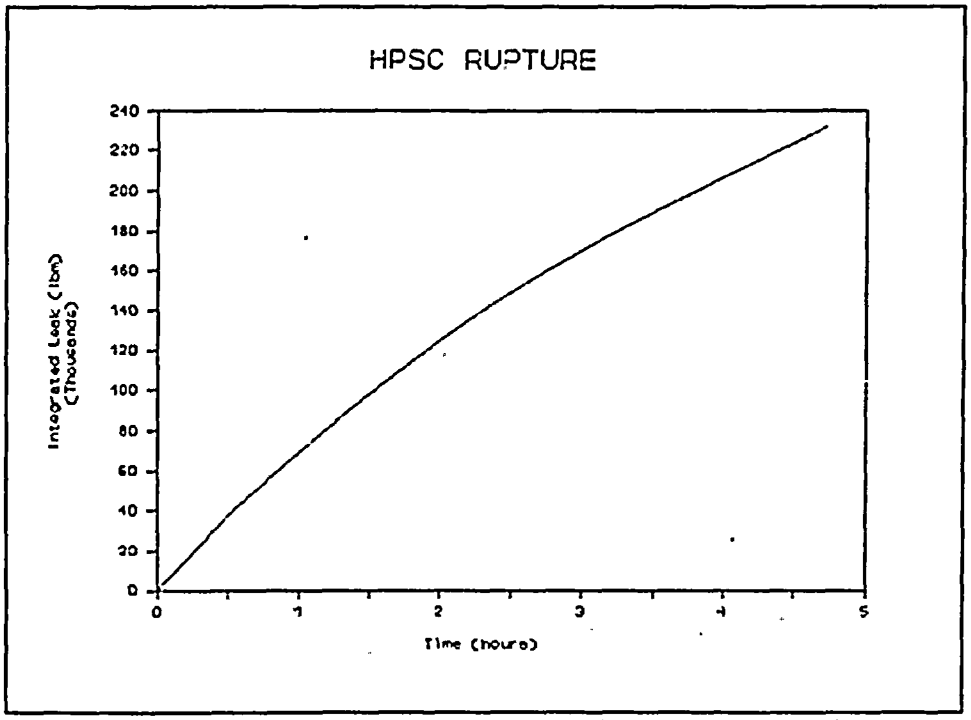

Figure 4-6 Integrated Reactor Coolant System Leakage

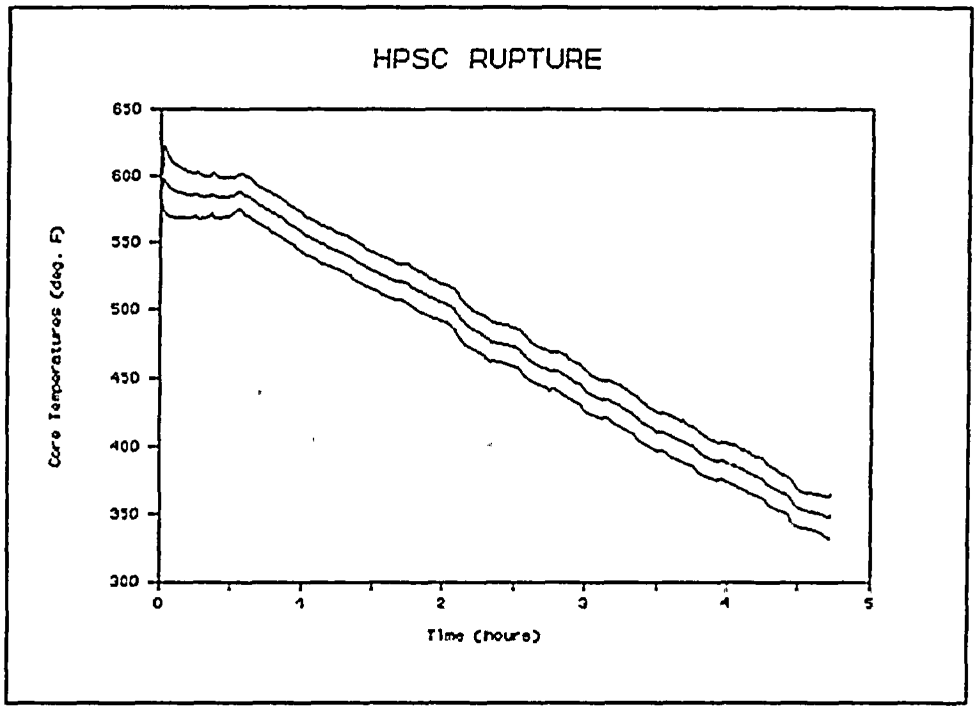

Figure 4-7 Reactor Coolant System Temperature
T_{hot}, T_{avg}, T_{cold}

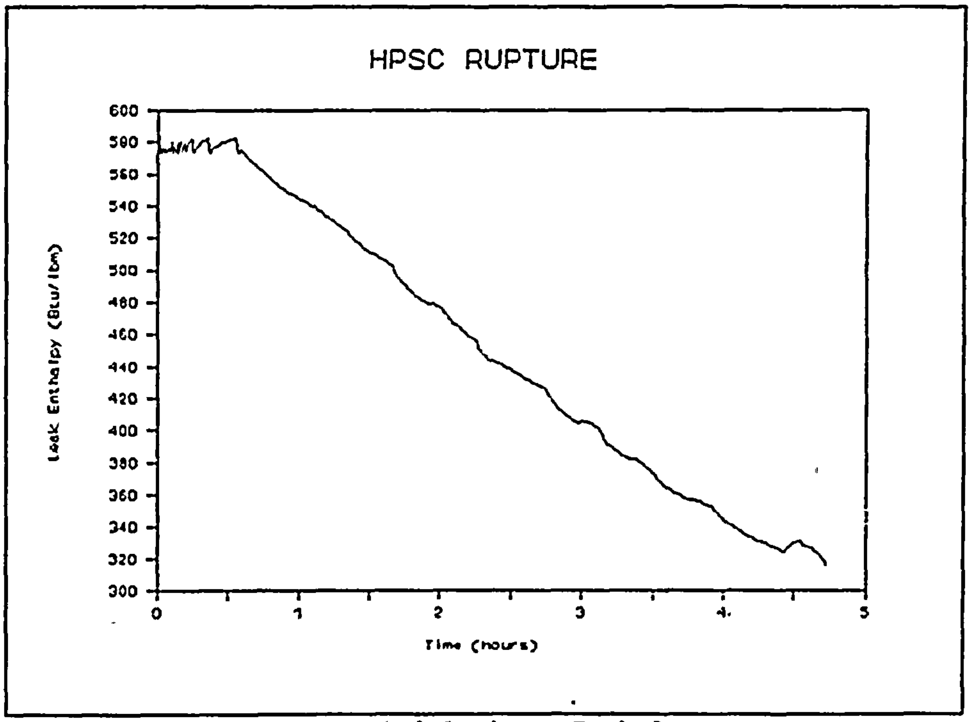

Figure 4-8 Leakage Enthalpy

B. Radiological Consequences

The radiological consequences of a postulated guillotine rupture of the high pressure seal cooler (at 100 percent rated power) with loss of off-site power was evaluated.

1. **Physical model** - The guillotine rupture of the high pressure seal cooler piping would result in degradation of the affected reactor coolant pump (RCP) and the nuclear cooling water system (NC). To maximize the radiological effects it is assumed that the operator would manually trip the reactor within 600 seconds of the initiating event, and that the Iodine Spike occurs coincident with the HPSC cooler rupture. The reactor trip would result in an automatic trip of the turbine which is assumed to cause a loss of off-site power. The limiting single failure for this scenario has been identified as loss of one diesel generator which would result in loss of one safety train.

Subsequent to the reactor trip the steam generator pressure will increase rapidly, resulting in steam discharge through the main steam safety valves. At the same time, the blowdown from the affected RCP high pressure seal cooler would pressurize the NC system. Pressurization of the NC system would result in rupture of the expansion tank located at the roof elevation of the auxiliary building and direct release to the environment.

After 2400 seconds the operator initiates a plant cool down at less than the technical specification cool down rate using steam generators, ADVs, auxiliary feedwater and the auxiliary spray system. It is assumed that the operators would be able to isolate the leakage from the RCS to the environment at 24 hours after the initiating event utilizing the NC system containment isolation valves.

2. **Major assumptions and conditions** - The following assumptions and parameters are used to determine radiation exposure at the site Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).

a. The accident doses are calculated for two different assumptions: a) assuming a Pre-accident Iodine Spike (PIS) and b) assuming a Generated Iodine Spike (GIS) at start of initiating event.

b. The Technical Specification limit of 1 uCi/gm Iodine-131 Dose Equivalent was used as initial maximum equilibrium RCS activity and a spiking factor of 500 employed for the Generated Iodine Spiking case.

c. The Technical Specification limit of 60 uCi/gm Iodine-131 Dose Equivalent was used to evaluate the Pre-accident Spiking case.

d. The contribution from noble gases are calculated based on 1% failed fuel, at the end of the equilibrium fuel cycle.

e. The postulated guillotine rupture of the high pressure seal cooler would not result in additional fuel failure.

f. The release rate (blowdown rate) is modeled as a time dependent parameter and is calculated based on the RCS behavior during the scenario.

g. Based on temperature and pressure response of the primary system and a constant enthalpy process, a time dependent flashing fraction was calculated and employed in both spiking cases.

h. The atmospheric dispersion factors used are based on Table 2.3-31 of PVNGS UFSAR.

i. RCS isolation would occur at 24 hours after initiation of the event.

3. **Mathematical Model** - The mathematical model employed to analyze the radiation release is based on NRC guidelines of the Standard Review Plan (SRP) 15.6.2, and Section 15.0.4 and Appendix 15B of the PVNGS UFSAR.

4. **Results** - The Two hour exclusion area boundary (EAB) and 8 hour low population zone (LPZ) Thyroid and whole body doses for both PIS and GIS are presented below. The Calculated EAB and LPZ doses are below 10 CFR 100 acceptance criteria.

LOCATION	Thyroid (rem)		Whole Body (rem)	
	PIS	GIS	PIS	GIS
EAB (0 - 2 hr)	257.1	153.2*	0.085	0.066*
LPZ (0 - 8 hr)	62.3	87.2	0.021	0.025

* These values are based on the existing Technical Specification limit for RCS activity of 1 uCi/gm I-131 DEQ. If RCS equilibrium I-131 DEQ activity is limited to 0.4 uCi/gm (the compensatory action limit), the 2 hour EAB dose to the thyroid would be 61.3 rem, less than 25% of the 10 CFR 100 limits.

5. **Conclusions** - Based on this evaluation, it can be concluded that with the compensatory action of limiting RCS equilibrium I-131 DEQ activity to 0.4 uCi/gm, the consequences of a postulated guillotine rupture of the seal cooler piping using SRP specified parameters are less than 25% of 10 CFR 100 limits.

VI. Compensatory Measures

To ensure that any leak of the seal cooler heat exchanger will be detected in a timely fashion and to meet the SRP off-site dose acceptance criteria in the event of a catastrophic failure of a HPSC, the following compensatory measures (Only applicable in Modes 1 through 4) have been put in place:

1. Chemistry procedures (74CH-9XC16, "Sampling and Analytical Schedule," and 74DP-9ZZ05, "Abnormal Occurrence Checklist") have been changed to provide for backup grab samples to be taken at least every 12 hours (not to exceed 14 hours) with RU-6 operable. This method will detect in-leakage lower than 0.08 gpm and also provides a confirmation of RU-6 operation. If RU-6 is out of service the backup samples will be taken at least every 4 hours (not to exceed 5 hours).

2. The radiation monitor RU-6 alarm response procedure (74RM-9EF41, "Radiation Monitoring System Alarm Response") and chemistry procedure (74DP-9ZZ05) have been modified to require specific actions be taken to quickly identify the source of the in-leakage to the NC system.

3. In the event manual sampling detects short lived fission product activity (indicative of Reactor Coolant Leakage in the NC system) or a radiation monitor alarm is received and manual sampling detects short lived fission product activity, an orderly plant shutdown to Mode 3 will commence within 4 hours, and the plant will be brought to Mode 5 within the following 30 hours. Sampling will continue during shutdown to monitor the leakage and to determine the source of the leakage.

If RCS in-leakage is determined not to be the source, the plant will not be shutdown and sampling shall continue to determine the source. Manual samples will be taken at least every 4 hours (not to exceed 5 hours) to ensure that no RCS leakage in the NC system would go undetected by the radiation monitor due to a possibly higher background activity.

4. PVNGS Unit 1 will monitor the primary coolant specific activity as required by technical specification 3/4.4.7. If the specific activity of the primary coolant is found to be greater than 0.4 uCi/gm I-131 DEQ for more than 48 consecutive hours, an orderly plant shutdown will be commenced and the unit will be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours. This compensatory action is implemented by changing the dose equivalent I-131 acceptance criteria of surveillance test procedure 74ST-9RC02, "Reactor Coolant System Specific Activity Surveillance Test" to 0.4 uCi/gm.

VII. Justification for Continued Operation

Continued operation of PVNGS until the implementation of a permanent design change to eliminate the possibility of HPSC pipe leak resulting in releases outside of containment is provided by the following analysis and compensatory actions.

The first barrier to such an occurrence is the high pressure piping in the HPSC. This piping is Schedule 80 austenitic stainless steel pipe with excellent erosion/corrosion resistance. The engineering analysis of the structure of the seal cooler determined that the stresses present in the piping are low enough that no mechanism could be identified which could cause or propagate a crack. Thus no failure of the piping would be anticipated. This conclusion was based upon a review of the original ASME Class I stress report(s) and supplemental stress evaluation for loading associated with design, normal, upset, emergency, faulted, and test conditions. The conclusion is further supported by the cumulative operating experience of pressurized water reactors. Only one previous seal cooler leak could be found during a review of the data bases for nuclear plant reliability data system, nuclear plant experience data base, and licensee event report data base. This event was not a catastrophic failure but a small leak and occurred at a Westinghouse plant in Switzerland during 1970. There have been no high pressure seal cooler leaks experienced at ABB/Combustion Engineering plants.

The radiation monitor (RU-6) in the NC system provides continuous monitoring of activity in the NC system and would detect this leak rate and alarm in the control room within one hour. As an additional precaution, routine samples of the NC system will be taken until a permanent design change is implemented. The routine sampling can detect leakage below 0.1 gpm. If the continuous radiation monitor becomes unavailable the sampling frequency will be increased to at least once every 4 hours (not to exceed 5 hours).

In the event that short half-life fission product activity (indicative of a leak from the RCS) is detected in the NC system, the affected unit will commence shutdown and cooldown within 4 hours unless it can be determined that the leak is from some other source which can be isolated. This action will ensure that should there be some flaw or unknown mechanism existing which could cause crack propagation in the cooler, further degradation would be arrested prior to piping rupture.

If a leak were to develop the operators could reasonably be expected to identify the affected pump either by seal cooler inlet temperature, which alarms in the control room, or by NC flow indications from the safety grade flow transmitters which have indicators in the control room. Following identification of the affected seal cooler the cooler could be isolated from the control room using the seal cooler inlet and outlet isolation valves. These valves are not safety grade but are designed to operate against full RCS differential pressure in order to isolate a seal cooler leak.

The double ended shear of HPSC piping was evaluated to address concerns over possible fuel failure by examining the spectrum of break sizes evaluated for a small break loss of coolant accident (LOCA). A failure of the RCP HPSC would correspond to a break size of 0.000975 ft^2 resulting in an initial leak rate of approximately 230 gpm. Additional leakage may occur as a result of RCP pump seal degradation due to loss of cooling and lubrication as a result of the HPSC failure. This leakage is expected to be less than 120 gpm with all three seal stages failed. Thus the total leakage from the RCS would be less than 350 gpm. The break sizes evaluated for the small break LOCA analyses were from 0.02 ft^2 to 0.5 ft^2. Break sizes in this range do not result in fuel failure. The 0.5 ft^2 break area (an approximate 56,150 gpm blowdown) bounds the expected leakage rate from both the HPSC rupture and RCP seal leakage due to loss of cooling. Based on this analysis, the high pressure seal cooler failure event would not result in fuel failure even if additional leakage from the RCP seals occurred.

An evaluation of the radiological consequences of the postulated catastrophic failure of a HPSC pipe was also performed. Using the extremely conservative Technical Specification limits for the RCS activity and the iodine spiking factor of the Standard Review Plan the evaluation results are well within the 10 CFR 100 limits. Considering the compensatory action of limiting RCS equilibrium I-131 DEQ activity to 0.4 uCi/gm, the GIS case off-site dose is reduced to less than 25% of 10 CFR 100 dose.

A probabilistic risk assessment (PRA) of the worst case scenario was performed. The PRA results indicate that during the next 4.5 months (anticipated time to Unit 1 refueling outage when modification will be implemented) the probability of having this event scenario at Palo Verde Unit 1 is 6.4E-6. This is considered a very low probability for non core melt sequences and the potential risks imposed on public health and safety during the JCO duration is, therefore, judged to be very minimal.

In summary the above analyses and actions demonstrate that: (1) the probability of a leak occurring in the HPSC is extremely small, (2) if in the unlikely event a HPSC leak did occur it would be detected quickly and the unit would commence a shutdown and cooldown within 4 hours, (3) operations personnel can be expected to determine the location of a leak and have the ability isolate the affected cooler effectively stopping the leak, (4) even if a catastrophic failure were to occur, with the compensatory action of prohibiting continued operation with equilibrium RCS activity levels above the administrative limit of 0.4 uCi/gm I-131 DEQ. and an iodine spiking factor of 500, the radiological consequence would be less than 25% of the 10 CFR 100 limits for the Generated Iodine Spiking case. Thus, the continued operation of Unit 1 does not adversely affect the health and safety of the public.

The implementation of corrective action is scheduled to be complete prior to start-up from the next refueling outage. This refueling outage is currently scheduled to start in February 1992. The corrective action being implemented is as follows:

> Installation of two redundant safety relief devices on the NC piping inside containment for purposes of mitigating the consequences of the LOCA by limiting the NC system pressure and ensuring RCS coolant is discharged to containment.