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SMALL AND LARGE BREAK CALCULATIONS FOR SAN ONOFRE UNIT 2

C. B. Davis
C. D. Fletcher
C. A. Dobbe

U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory

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Author(s): C. B. Davis
C. D. Fletcher
C. A. Dobbe

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EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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ABSTRACT

Analyses were performed in support of the Nuclear Regulatory Commission's licensing of the San Onofre Unit 2 Combustion Engineering pressurized water reactor for low power operation. The RELAP4/MOD7 computer code was used to calculate the effects of a small and a large cold leg break on San Onofre Unit 2 while operating at 5% rated power. Results of the calculations were used to provide boundary and initial conditions for subsequent core boiloff calculations performed by the Nuclear Regulatory Commission.

Summary

Analyses were performed in support of the Nuclear Regulatory Commission's licensing of the San Onofre Unit 2 Combustion Engineering pressurized water reactor (PWR) for low power operation. The RELAP4/MOD7 computer code was used to calculate the effects of a small and a large cold leg break on San Onofre Unit 2 while operating at 5% rated power. A complete failure of all pumped Emergency Core Coolant (ECC) systems was assumed. Results of the RELAP4/MOD7 calculations were used to provide boundary and initial conditions for subsequent core boiloff and damage calculations performed by the Nuclear Regulatory Commission. The purpose of the core boiloff calculations was to determine the time of core damage.

The small break calculation considered a 0.0508 m diameter hole in a cold leg pipe. The calculation represented nearly 4 hours of the small break transient. No heatup of the reactor fuel was calculated although the core nearly uncovered prior to accumulator injection. At the time the calculation was terminated, the accumulators were empty and the upper plenum mixture level was 0.09 m below the bottom of the hot leg nozzles and was decreasing slowly. Hand calculations indicated that the core would not uncover for at least 2 additional hours.

The large break calculation considered a 200% double-ended offset shear of a cold leg pipe. The calculation represented 83 s of the large break transient. No significant heatup of the reactor fuel was calculated. The reactor vessel voided during blowdown and was refilled by the accumulators, which emptied at 80 s. The effect of the stored energy in the steam generators and the reactor vessel wall was significant during the transient. The energy transferred to the primary coolant from the steam generators and the reactor vessel wall was about 20 times higher than the core power at the end of the calculation. Accumulator temperature also had a significant effect upon the calculated refill of the vessel. With cold accumulator water, 80% more liquid was present in the vessel at 80 s compared to a calculation with artificially warm accumulator water.

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1. INTRODUCTION

Two calculations were performed in support of the Nuclear Regulatory Commission's licensing of the San Onofre, Unit 2 Combustion Engineering pressurized water reactor (PWR) for low power operation.

Thermal-hydraulic plant conditions were simulated using the RELAP4/MOD7 computer code[1] and a model of the San Onofre PWR which was derived from an existing model of the Combustion Engineering Calvert Cliffs, Unit 1 PWR. The plants are similar in some respects (such as primary volume and pipe sizes) and different in others (such as rated core power, and core bundle and steam generator configuration.) The existing Calvert Cliffs model was modified to account for these differences and a description of the changes made is included in this report. Since the primary volume to reactor coolant piping flow area ratios are the same for two plants, results obtained using the modified Calvert Cliffs model for any size break are representative of those that would be expected using an exact model of the San Onofre plant if one were available.

Plant operating conditions at 5% of full power were simulated and the effects of a small and a large cold leg break were calculated using best estimate assumptions except as noted. The small break calculation considered a 0.0508 m diameter hole in a cold leg pipe and the large break calculation considered a 200% offset shear of a cold leg pipe. In both calculations a complete failure of all pumped emergency core coolant (ECC) (charging, high pressure injection, and low pressure injection) was assumed.

Results of the calculations were used to provide boundary and initial conditions for subsequent core boiloff and damage calculations performed by the Nuclear Regulatory Commission using another computer code.

This report describes the model used in Section 2. Descriptions of the changes made in converting the Calvert Cliffs model to a San Onofre Model (2.1), nodalization (2.2), code options (2.3), initial conditions (2.4), and boundary conditions (2.5) are included. The results of the small and large break calculations are described respectively in Sections 3.1 and 3.2. Conclusions are summarized in Section 4 and references given in Section 5.

2. RELAP4 MODEL DESCRIPTIONS

The RELAP4 models used to make the San Onofre small and large break calculations are described in this section. The San Onofre small and large break calculations were performed with models based on the Calvert Cliffs model described in Reference 2. The changes made to the Calvert Cliffs model to represent San Onofre are described below. Descriptions of the nodalizations, code options, and initial and boundary conditions are also presented. Input decks for the calculations are stored under configuration control number F00506 at the Idaho National Engineering Laboratory (INEL). The computer code, version G101 of RELAP4/MOD7, is stored under configuration control numbers F00509 (main coding) and H009982B (steam tables) at INEL.

2.1 Modifications for San Onofre Configuration

Several changes were made to the Calvert Cliffs model to represent the San Onofre configuration of the vessel, steam generators, and accumulator components.

The model of the vessel was modified primarily to account for a different fuel geometry. The original Calvert Cliffs model represented 14 x 14 fuel bundles while San Onofre contains 16 x 16 fuel bundles. The core heat slabs were modified to represent the geometry of the San Onofre 16 x 16 fuel bundles. The hydraulic geometry of the core was not altered and remained in the Calvert Cliffs configuration. In addition, the core bypass flow rates were modified to be consistent with the bypass flow rates presented in the San Onofre Final Safety Analysis Report[3] (FSAR). The core bypass flow from the downcomer to the upper head corresponded to 0.7% of the total vessel flow and represented the combined leakage through the upper head and the leakage due to the hot leg nozzle clearances.

The steam generator heat removal capacity of the San Onofre PWR is greater than that of the Calvert Cliffs PWR. While the number and size of tubes is the same, the average tube length is longer (18.44 vs 14.63 m) which results in a larger heat transfer area. The model was revised to account for this difference by increasing the height of the uppermost volumes and heat slabs and the secondary mixture levels in each steam generator.

The San Onofre accumulators[a] contain 60% more liquid and are operated at 2.76 MPa higher pressure than the Calvert Cliffs accumulators. Best-estimate values for the San Onofre accumulator liquid volume, gas volume, and pressure were used in the small and large break calculations.

2.2 Nodalization

2.2.1 Small Break Nodalization

The nodalization used for the small break analysis is shown in Figure 1. This nodalization is the same as the nodalization used in the Combustion Engineering audit calculations of open power operated relief valve transients[2] except that junctions for the power operated relief valve and charging, high pressure, and low pressure injection were removed and the pressurizer was modeled by a single volume.

The core was modeled with three volumes and six heat slabs which represented the fuel rods. Each steam generator was modeled with four primary volumes and heat slabs and a single secondary volume. One loop was modeled using a single hot leg and cold leg while the other loop was prototypically modeled using a single hot leg and two cold legs. The accumulators were connected to the downcomer upper annulus to avoid atypical primary system depressurization during injection of cold accumulator water. Additional heat slabs were located on the downcomer, core bypass, lower plenum, upper plenum, and upper head.

2.2.2 Large Break Nodalization

The nodalization of the RELAP4 model used to perform the large break calculation is shown in Figure 2. The large break nodalization was similar

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a. The word "accumulator" is used to represent the safety injection tank.

to that for the small break except that one pump discharge leg was renodalized to better represent a 200% double-ended offset shear break and the accumulators were connected to the cold legs to allow a more physical representation of the bypass of the ECC. The accumulator connected to the broken pump discharge leg was assumed to flow directly into the containment and was not modeled.

2.3. Code Options

Different code options were applied in the small and large break calculations because different phenomena were expected in the two transients. The code options used in the small break calculation were based on experience obtained at INEL in applying RELAP4/MOD7 to small break transients. The code options used in the large break calculation were based on the user guidelines derived during the assessment[4] of the RELAP4/MOD6 computer code.[5] The options used for each calculation are described below.

2.3.1 Code Options for the Small Break Calculation

The following user-selected code input options were used for the small break calculation. Critical flow was modeled using the Henry-Fauske/Homogeneous Equilibrium Model option. A multiplier of 1.0 was applied to both models. The vertical phase slip model was applied in the downcomer and core (Junctions 1 and 3, and 5 through 10, see Figure 1). The new slip model, which uses a flow-regime dependent slip velocity correlation, was applied. The Wilson bubble rise model was used in the pressurizer, loop seals, and the vessel. The lower plenum, core, and upper plenum were modeled as a vertical stack. The downcomer was also modeled as a vertical stack. A bubble gradient of 0.8 was used with the Wilson bubble rise model. The parameters of the bubble rise model used in the steam generator secondaries were calculated by the self-initialization feature. To save calculational time, the bubble rise models in the upper head and pressurizer were tripped off and homogeneous calculations made after the respective volumes were depleted of liquid. The default and/or recommended heat transfer options of the RELAP4/MOD6 computer code were used.

Specifically, the HTS2 heat transfer surface was employed. CHF was calculated with the W-3 correlation in the subcooled regime, Hsu and Beckner's modified W-3 correlation in the saturated high flow regime, and Smith and Griffith's modified Zuber correlation in the saturated low flow regime. Transition boiling was calculated with the modified Tong-Young correlation. Film boiling was calculated with the Condie-Bengston III correlation. The natural convection heat transfer option was used in both steam generator secondaries. The self-initialization feature was used to calculate initial system pressure and temperature distributions. The enthalpy transport model was used for initialization but was not used during the transient. The compressible form of the momentum equation (MVMIX = 0) was used at all junctions except that the incompressible momentum equation (MVMIX = 3) was used at the pressurizer to hot leg, accumulator to inlet annulus, and fill junctions and those junctions connecting the vessel with the hot or cold legs. The alternate (recommended) water packing and liquid mass depletion model was applied.

Core power was calculated with the RELAP4/MOD7 reactor kinetics model. Fission power was calculated with point kinetic equations. Reactivity effects due to scram and feedback effects due to moderator temperature, moderator density, and fuel temperature were modeled. The reactivity curves input to the model were based on information presented in the San Onofre FSAR. Decay heat, including the effect of heavy element decay, was calculated assuming an infinite operating period.

2.3.2 Code Options for the Large Break Calculation

The code options used in the large break calculations were similar to those used in the small break calculation except as described below.

Phase separation in the reactor vessel was calculated with the vertical phase slip model which was applied at the same junctions as in the small break calculation. The Wilson bubble rise model was applied only in the pressurizer and upper head.

Critical flow was calculated using stagnation critical flow properties. The stagnation properties option was tripped off when the upper plenum pressure dropped below 0.34 MPa

The polytropic air expansion model was applied in the accumulator volumes. A polytropic exponent of 1.4 was used.

The nonequilibrium model was applied in the pump discharge legs and reactor vessel. Default values of model parameters were used except that, as recommended, the nonequilibrium model was tripped off in those volumes with a quality below 1.0E-6 and the model reactivated in these volumes when the void fraction exceeded 0.25.

The first water packing model, the model which was available in RELAP4/MOD6, was used in the large break calculation.

2.4. Initial Conditions

The small and large break transients were assumed to start with the plant operating at 5% rated power and 100% core flow. The basic initial conditions which were input to the computer model are shown in Table 1. The initial conditions represent a best-estimate of the actual operating status of the plant at 5% power. The basic initial conditions were obtained, when available, from the San Onofre FSAR. The self initialization feature of RELAP4/MOD7 was use to calculate the steady-state steam generator energy balance and the thermodynamic state of each control volume. Specific assumptions pertaining to the initial conditions are described below.

The initial pressurizer liquid volume was reduced to 50% of the normal volume at 100% power to approximate the effect of the pressurizer level control system on the pressurizer liquid inventory.

The average of the vessel inlet and outlet fluid temperatures was calculated by interpolating between the average fluid temperatures presented in the FSAR for 100% power and 0% power conditions. The average

TABLE 1. INITIAL CONDITIONS FOR THE SMALL AND LARGE BREAK CALCULATIONS

Parameter	Initial Value
Core Power, MW	169.5
Total Vessel Flow, kg/s	18648.
Pressurizer Pressure, MPa	15.51
Vessel Inlet Fluid Temperature, K	558.4
Vessel Outlet Fluid Temperature, K	560.1
Core Bypass Mass Flow, kg/s	261.1
Upper Head Bypass Mass Flow, kg/s	130.5
Pressurizer (Including Surge Line)	
Liquid Volume, m^3	12.1
Vapor Volume, m^3	31.1
Steam Generator	
Feedwater Flow, kg/s/generator	55
Feedwater Enthalpy, J/kg	9.881×10^5
Secondary Pressure, MPa	6.62
Secondary Liquid Volume, m^3	94.38
Secondary Vapor Volume, m^3	132.33
Accumulator	
Liquid Volume, m^3/tank	49.36
Gas Volume, m^3/tank	14.36
Pressure, MPa	4.31
Temperature, K	322
Average Fuel Centerline Temperature, K	589.5

centerline fuel temperature was calculated by interpolating between the fuel-to-coolant differential temperatures presented in the FSAR for 100% and 0% power.

The axial power factors used for the core heat slabs are shown in Table 2. The power factors represent an axial power profile obtained from the FSAR and are applicable for 35% power early in core life.

2.5. Boundary Conditions

The boundary conditions applied to the small and large break calculations included steam generator feedwater and steam mass flow rates, auxiliary feedwater mass flow rates, and containment pressure. The response of most of these boundary conditions during the transients was related to the occurrence of the scram signal. The scram signal was generated on a low pressurizer pressure for these transients. The low pressure set point for San Onofre was 11.03 MPa.

The occurrence of the scram signal actuated the reactor scram; time delays were built into scram reactivity vs. time curve to account for solenoid deenergization and control rod drop time. The primary coolant pumps were tripped off when the reactor scram signal occurred. The steam generator feedwater and steam lines were closed linearly between the time the scram signal occurred and 5 s after the scram signal occurred. Auxiliary feedwater was initiated 30. s after the occurrence of the scram signal. The auxiliary feedwater flow rate was modeled as a function of secondary mixture level. The auxiliary feedwater flow attempted to control secondary mixture level between 13.4 m and 14.0 m above the top of the tube sheet. An auxiliary feedwater flow rate of 43.5 kg/s/generator was applied when the secondary mixture level was below 13.4 m. No auxiliary feedwater flow was permitted when the level exceeded 14.0 m. Auxiliary feedwater temperature was 322 K. The accumulators were the only source of ECC during the transients.

The containment was modeled as a time-dependent volume for both transients. For the small break transient, a constant containment pressure of 0.101 MPa was assumed. The actual magnitude of the assumed containment pressure was not important because the break remained choked in the small break calculation. For the large break transient, the containment was assumed to be filled with saturated steam. The containment pressure history was taken from the San Onofre FSAR and represented the response calculated for a 200% cold leg break from 100% power.

TABLE 2. AXIAL POWER FACTORS

TOP of Core
0.0946
0.1655
0.1956
0.2031
0.1956
0.1456
Bottom of core

3. RESULTS

3.1. Results of the Small Break Calculation

The times at which significant events occurred in the calculation are listed in Table 3.

At time zero a 0.0508 m diameter break was opened in the cold leg. The pressurizer emptied at 40 s and at 45.3 s the low pressurizer pressure scram setpoint was reached causing the reactor scram, reactor coolant pump trip, and termination of steam generator main feedwater and steam flows. The primary pressure rapidly fell until, at about 270 s, it was slightly above the secondary pressure as shown in Figure 3.

The secondary mixture level, which had fallen as a result of reactor scram and secondary isolation, recovered to the normal level shortly after initiation of auxiliary feedwater at 75.3 s as shown in Figure 4. The secondary level was maintained at the normal level for the remainder of the transient by throttling of auxiliary feedwater.

The reactor vessel upper head drained at 390 s and the upper plenum level (Figure 5) was only 0.1 m above the core when accumulator injection began at 4158 s. Figure 6 shows the accumulators injected liquid in stages until their inventory was depleted at 7150 s. Injection in stages was caused by repressurizations of the primary system above the accumulator pressure following surges in accumulator injection. Accumulator injection caused a refilling of the vessel as noted by the upper plenum level, and a depressurization of the primary and secondary systems. Following accumulator depletion, the system was stabilized with a primary pressure of 862 kPa, a secondary pressure of 1930 kPa, and the upper plenum mixture level near the bottom of the hot leg nozzles.

The quality in the volume with the break, shown in Figure 7, increased to about 0.9, returned to zero during accumulator injection, then increased to nearly 1.0 following accumulator depletion.

TABLE 3. CHRONOLOGY FOR THE SMALL COLD LEG BREAK CALCULATION

Time (s)	Event
0.0	Blowdown initiated
40.0	Pressurizer emptied
45.3	Scram signal generated
46.0	Reactivity insertion due to scram initiated
75.3	Auxiliary feedwater initiated
390	Upper head drained
4158	Accumulator injection initiated
7150	Accumulators emptied
12514	Calculation terminated

The calculation was terminated at 12514 s. At that time, the primary pressure was 738 kPa and decreasing at a rate of about 0.07 kPa/s. The secondary pressures were 1980 kPa in the broken loop and 1430 kPa in the intact loop. The upper plenum mixture level was 1.08 m which is 0.09 m below the bottom of the hot leg nozzles. The break mass flow rate was 2.19 kg/s and the quality of the break flow was 0.978. 12,400 kg of liquid remained in the vessel above the core. Steam production was being relieved primarily by two paths: through the bypass from the upper head to the upper annulus and through the intact loop which by this time had been almost completely voided. The model included a broken loop which contained two cold legs. In the two broken loop cold legs, significant amounts of liquid were still present: 3500 kg in the intact cold leg and 3000 kg in the broken cold leg. At the problem termination time the core power was 1.786 MW, heat addition from the steam generator secondaries was approximately 10% of core power, and heat addition from other heat structures was approximately 30% of core power.

Analysis indicates that the primary pressure and break mass flow rate would continue to decrease until the energy removal rate at the break equaled the decay heat production rate. Using conditions at 689 kPa, the mass flow rate at equilibrium will be:

$$\frac{1.786\ \text{MW}}{0.293\ \frac{\text{W}}{\text{Btu/HR}}\ 3600\ \frac{\text{sec}}{\text{HR}}\ 889\ \frac{\text{BTU}}{\text{LBM}}} = 1.90\ \frac{\text{LBM}}{\text{sec}} = 0.862\ \text{kg/s}$$

This corresponds to a break mass flux of 3.679 kg/s/m^2 at equilibrium. This break mass flux is associated with an equilibrium primary pressure of 285 kPa. Thus the equilibrium conditions expected are a break steam flow of 0.862 kg/s at a primary pressure of 285 kPa.

Considering only the liquid in the vessel and the break mass flow rate at problem termination, core uncovery could be expected at

$$12514 + \frac{12400}{2.19} = 18178\ \text{s}$$

This represents a lower bound estimate of core uncovery time since it assumes none of the 6530 kg of liquid in the loops at problem termination time migrates to the vessel and the break mass flow rate does not decrease.

Considering only the liquid in the vessel and the equilibrium break mass flow rate. The amount of liquid expected to flash in depressurizng from 689 to 276 kPa is

$$\frac{12{,}400 \text{ kg} \times 145.17 \frac{\text{kJ}}{\text{kg}}}{2068 \frac{\text{kJ}}{\text{kg}}} = 871 \text{ kg}$$

and the core uncovery could be expected at

$$12514 + \frac{12400-871}{.862} = 25873 \text{ s}$$

These two estimates represent a range of the core uncovery time. Both extremes of the range would be increased if credit were taken for any of the remaining liquid in the loops.

3.2. Results of the Large Break Calculation

The times at which significant events occurred in the calculation are listed in Table 4.

Calculated pressures for the upper plenum, intact loop steam generator, and intact loop accumulator are shown in Figure 8. The upper plenum pressure dropped almost instantly from its initial value of 15,500 kPa to 6,760 kPa which was slightly lower than the saturation pressure corresponding to the initial hot leg temperature. The primary coolant temperature decreased slightly during the sudden decompression. The calculated pressure was similar to that expected in an isothermal blowdown because the plant was assumed to be operating at a small (2 K) hot leg to cold leg temperature differential at the start of blowdown. The upper plenum pressure then decreased at a more gradual rate and reached

TABLE 4. CHRONOLOGY FOR THE 200% COLD LEG BREAK CALCULATION

Time (s)	Event
0.0	Blowdown initiated
4.6	Pressurizer emptied
5.3	Scram signal generated
6.0	Reactivity insertion due to scram initiated
13	Fuel rods dried out
13.3	Accumulator flow initiated
22	Minimum vessel coolant inventory occurred
35	Lower plenum refilled
35.3	Auxiliary feedwater initiated
52	Fuel rods quenched
80	Accumulators emptied
82.9	Calculation terminated

containment pressure near 24 s. The pressure in the intact loop steam generator secondary, which was representative of both steam generators, decreased 30% during the calculation. Most of the pressure decrease occurred after 35 s and was due to heat transfer from the steam generator to the primary coolant, and to a lesser extent, the injection of cold auxiliary feedwater. The pressure in the intact loop accumulator, which was representative of the accumulators in both loops, began decreasing at 13 s when the pressure of the primary coolant dropped below the nitrogen cover pressure allowing ECC to flow into the primary system. The accumulator pressure decreased smoothly until the tank emptied at 80 s when a valve was closed to prevent air from entering the primary coolant system. RELAP4/MOD7 does not accurately represent air flow.

The total fluid mass in the reactor vessel, divided by the initial mass, 9.689×10^4 kg, is shown in Figure 9. The vessel coolant inventory decreased rapidly as mass exited through the large cold leg break. Flow from the accumulators initiated at 13 s but most of the ECC water was bypassed out the break until 22 s when the vessel was essentially voided with an average void fraction of 0.99. By 24 s the pressure of the primary coolant dropped below that of the containment which resulted in backflow from the containment and the end of ECC bypass. The reflood of the core was initiated at 35 s when the lower plenum was refilled with liquid. The core was completely reflooded at 52 s. The vessel was almost full of subcooled liquid at 80 s, when the accumulators emptied, although a two-phase fluid remained in the upper head and control element assembly regions (Volumes 12 and 11, respectively, Figure 2). The vessel contained more mass at the end of the calculation than at the start because the fluid was cooler and denser.

The calculated fuel rod cladding temperature and corresponding fluid temperature for the uppermost core heat slab are shown in Figure 10. The fluid temperature generally represented saturation temperature except after 52 s when the core fluid was subcooled. The calculated cladding temperature shown in Figure 10 was representative of the entire core. The calculated cladding temperature closely followed the fluid temperature until 13 s when the fuel rod dried out. The fuel rod temperature then departed from the fluid temperature although cooling of the fuel generally continued. The lower plenum was refilled at 35 s initiating core reflood which caused a sudden decrease in cladding temperature of at least 40 K throughout the core. The core was reflooded and the fuel rods completely quenched by 52 s.

The power generated in the core due to fission plus radioactive decay had a relatively small effect upon the transient as illustrated by Figure 11 which compares the core power with the total power transferred to the primary coolant from the steam generators, fuel rods, and passive vessel heat structures. The total power transferred to the primary coolant was generally between one and two orders of magnitude higher than the core power which illustrates the importance of stored energy in the steam generators, fuel rods, and vessel heat structures for this transient. The relative unimportance of the core power is not a surprising result since the calculation was initiated at only 5% rated power and hence nearly represented an isothermal blowdown. At the end of the calculation, the steam generators and the reactor vessel wall were each transferring about 10 times more power into the primary coolant than the decay power. Hand calculations indicated that the vessel wall would take about 10 minutes to cool down to the primary coolant temperature while the steam generators would take about 25 minutes to cool down. The energy stored in these components was equivalent to about 350 minutes of decay power. The stored energy in these components must be modeled well to perform accurate core boiloff calculations.

Most of the total power transferred to the primary system, see Figure 11, was from different components at different times during the calculation. For example, most of the heat transfer between 6 s and 20 s was from the steam generators. The voiding of the primary coolant system thermally isolated the steam generators which reduced the total power to the primary coolant until core reflood was initiated at 35 s. The subsequent removal of stored heat from the fuel caused a large increase in power transferred to the primary coolant. The reflooding of the core also resulted in the entrainment of a small amount of liquid from the core to the steam generators where the liquid was evaporated resulting in significant heat transfer to the primary system. The steam generator heat transfer was limited by the amount of liquid entrainment. Consequently, the calculated liquid entrainment is a significant parameter in determining the cooldown rate of the steam generators. The heat transfer from the fuel rods decreased after the completion of core reflood at 52 s. At the end of the calculation, most of the heat transfer to the primary coolant was from the steam generators and the reactor vessel wall.

The calculation was terminated at 82.9 s which was 2.9 s after the accumulators emptied of liquid. The state of the primary coolant system at the end of the calculation is described below. The downcomer, lower plenum, core, and upper plenum were filled with subcooled liquid whereas two-phase mixture remained in the upper head and control element assembly regions. The hot legs and pump discharge leg of each unbroken loop were also filled with liquid. The pump suction legs and steam generator U-tubes were mostly filled with steam. The fuel rods were quenched. The steam generators and vessel heat slabs were contributing more power to the primary coolant system than the reactor fuel. The calculated refilling of the vessel above the cold leg and hot leg nozzles may be unrealistic since the vessel level would probably not rise above the top of the nozzles in an accident. A more detailed nodalization in the vicinity of the nozzles and/or the use of a bubble rise model to track vessel levels would be required to more accurately represent fluid behavior above the nozzles during the reflooding of the vessel.

A preliminary calculation was performed with warm water in the accumulators to determine the effect of ECC temperature on the response of the primary coolant system. This preliminary calculation was identical to the calculation described above except that the initial ECC temperature was 372 K instead of 322 K and a different water packing model was used as described later. The ECC temperature had a significant effect upon the calculated vessel fluid inventory as shown by Figure 12. The reactor vessel refilled faster and less ECC water bypassed out the break with cold ECC water. Nearly 80% more liquid was in the vessel at the time the accumulators emptied in the calculation with cold ECC water. This additional liquid in the vessel would require more than 10 hours to be boiled away by decay heat.

Figure 13 shows the effect of ECC temperature on upper plenum pressure which explains the effect of ECC temperature on vessel inventory shown in Figure 12. The injection of cold ECC caused more condensation which resulted in a lower primary coolant pressure compared to the calculation with warm ECC. When the primary coolant pressure was below containment pressure, such as between 25 s and 35 s, steam flowed from the containment to the downcomer which prevented ECC bypass and promoted vessel refill. The initiation of core reflood at 35 s and subsequent removal of stored energy from the fuel rods and steam generators produced steam which pressurized the primary coolant. In the calculation with cold ECC water, the primary coolant pressure again decreased below containment pressure shortly after the core was quenched at 52 s which prevented any further ECC bypass and promoted vessel refill. In the calculation with warm ECC, less condensation occurred which allowed the heat transfer from the fuel rods and steam generators to maintain the primary coolant pressure above containment pressure which promoted ECC bypass and even resulted in a reduction of vessel inventory. The major difference between calculations was that less condensation occurred with warm ECC which allowed the primary coolant pressure to generally exceed containment pressure thus promoting ECC bypass.

RELAP4/MOD7 has two water packing models. One model, called the first water packing model, was available in earlier versions of the code such as RELAP4/MOD6. An alternate water packing model was developed and recommended for use in RELAP4/MOD7. The computer run time was found to be sensitive to the selection of the water packing model although the calculated results were not sensitive to model selection. The calculation with the cold (322 K) ECC water was run with the first water packing model and averaged 180 cp s per real s. The calculation with the warm (372 K) ECC water was run with the alternate water packing model before 59 s and with the first model after 59 s. The warm ECC water calculation averaged 1000 cp s per real s between 45 and 58 s and 60 cp s per real s after 59 s. Changing from the alternate to the first water packing model improved the computer run time by nearly a factor of 20. The calculation with the alternate water packing model ran slowly because the time step control algorithim selected the input minimum time step of 2.5×10^{-5} s to satisfy the saturation line crossing criterion as a result of liquid filling control volumes. Discussions with code development personnel indicate that the use of the alternate water packing model may be inconsistent with the code's time step control algorithim for some transients. The alternate water packing model probably could have been used economically with a large minimum time step.

4. CONCLUSIONS

The following conclusions were derived as a result of the calculations.

1. For the small break, core uncovery time estimates, assuming no remaining loop liquid migrates to the vessel, were 18178 s using the break flow rate at problem termination time, and 25873 s using the expected equilibrium break flow rate.

The small break calculation was terminated at 12,514 seconds for economic reasons and estimates of core uncovery time were made by hand calculation. Both of the above estimates would be increased if some of the remaining loop liquid was assumed to migrate to the vessel.

2. The stored energy in the heat structures and steam generator secondaries was important in the large and small break calculations because of the low initial core power.

At the end of the small break calculation heat addition from the heat structures was approximately 30% of the core power and heat addition from the secondaries was approximately 10% of core power. At the end of the large break calculation, the power deposited in the primary coolant from the steam generators and the reactor vessel wall was about 20 times larger than the core decay power (see Figure 11.). The remaining stored energy in the steam generators and reactor vessel was equivalent to about 6 hours of decay power. The stored energy in these components must be modeled well to perform accurate core boiloff calculations.

3. ECC temperature had a significant effect upon the ECC bypass and vessel refill results for the large break.

Calculations performed with cold (322 K) and warm (372 K) ECC water showed that the condensation caused by cold ECC water affected the differential pressure between the primary coolant and the containment (see Figure 13) which reduced ECC bypass resulting in substantially more liquid in the vessel when the accumulators emptied. The additional liquid in the vessel with the cold ECC would require more than 10 hours to be boiled away by decay heat.

4. <u>A detailed nodalization and/or the use of a bubble rise model may be necessary to accurately track vessel mixture levels once the levels approach the cold leg and hot leg nozzles during the refill of the vessel.</u>

The downcomer and upper plenum refilled completely in the large break calculation although in reality the vessel would probably not refill above the nozzles.

The following conclusion is applicable to the usage of RELAP4/MOD7.

1. <u>The large break calculation computer run time was sensitive to the selection of water packing model.</u>

As described in Section III.3, computer run time for the large break calculation was improved by a factor of 20 when the first water packing model was used instead of the alternate water packing model. The alternate water packing model, which was developed for use with RELAP4/MOD7, may not be consistent with the code's time step control algorithim for some transients. The economical use of the alternate water packing model may require the use of large minimum time step sizes.

5. REFERENCES

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2. C. B. Davis and C. D. Fletcher, Audit Calculations of Open Power Operated Relief Valves in a Combustion Engineering PWR, EGG-CAAP-5054, November 1979.

3. Final Safety Analysis Report, San Onofre Nuclear Generating Station Units 2&3, Southern California Edison Company and San Diego Gas and Electric Company.

4. Code Assessment Branch, Assessment of the RELAP4/MOD6 Thermal-Hydraulic Transient Code for PWR Experimental Applications, EGG Idaho Inc., CAAP-TR-78-035, December 1978.

5. RELAP4/MOD6 A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems User's Manual, EG&G Idaho, Inc., CDAP TR 003, January 1978.

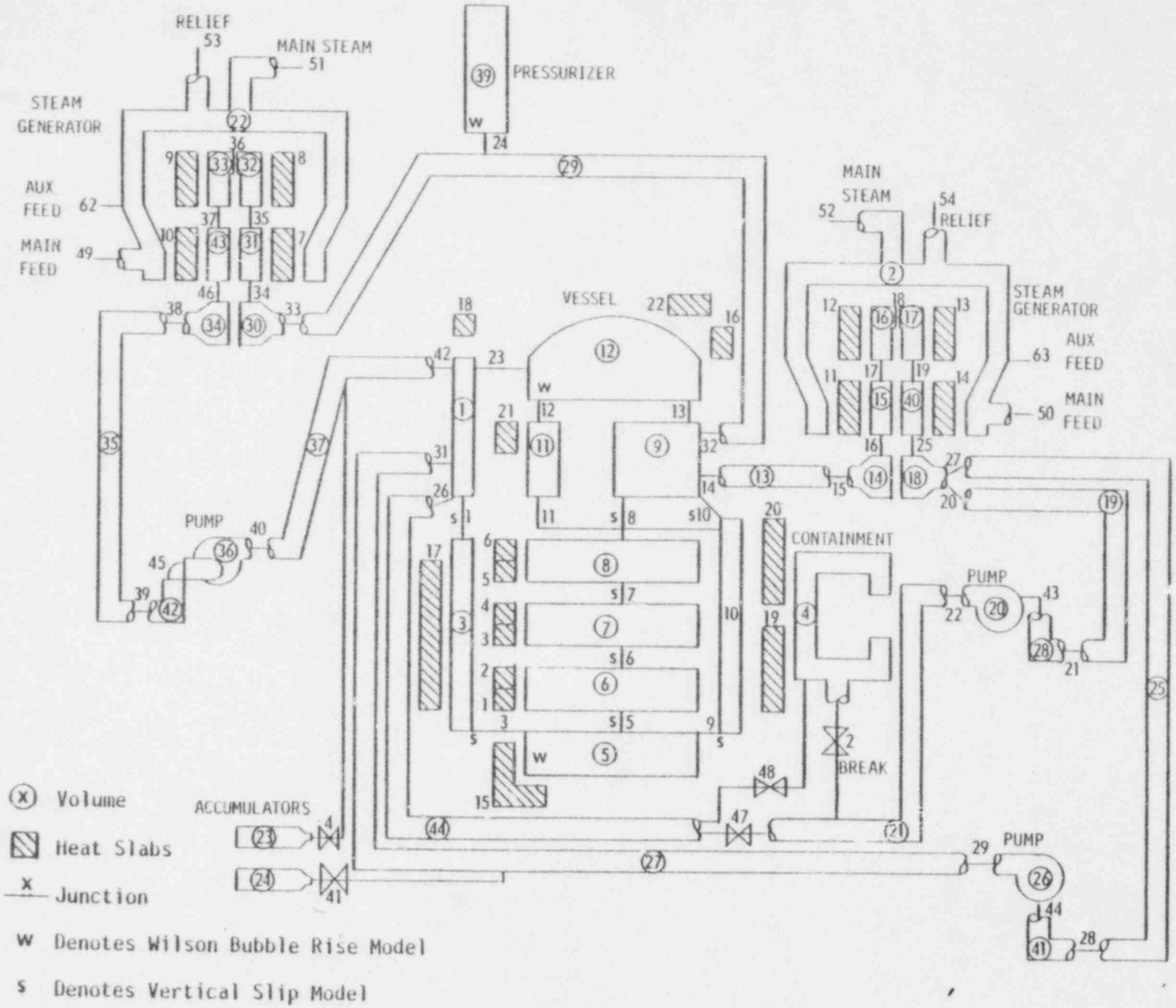

Figure 1. Nodalization for the small break calculation.

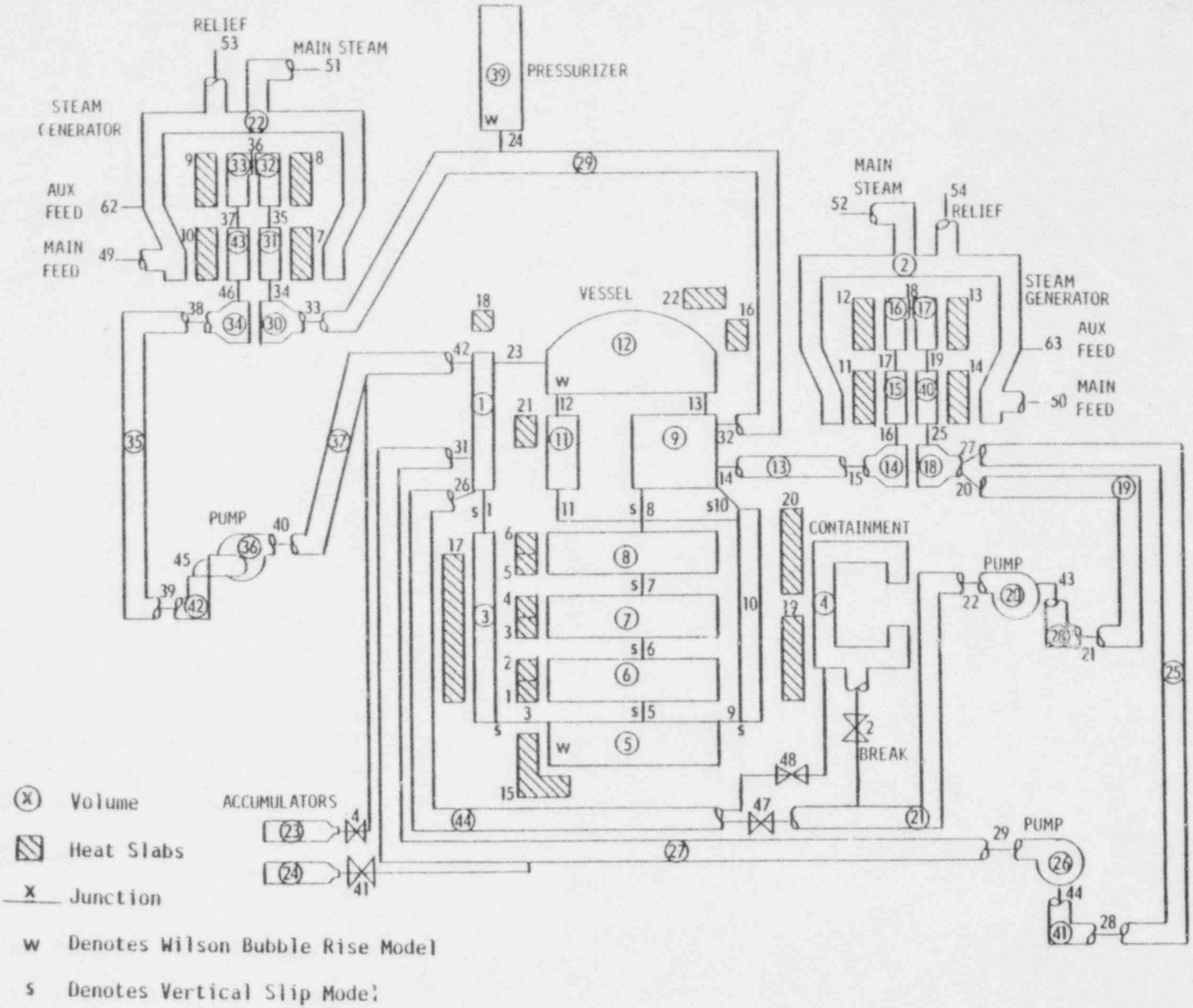

Figure 2. Nodalization for the large break calculation.

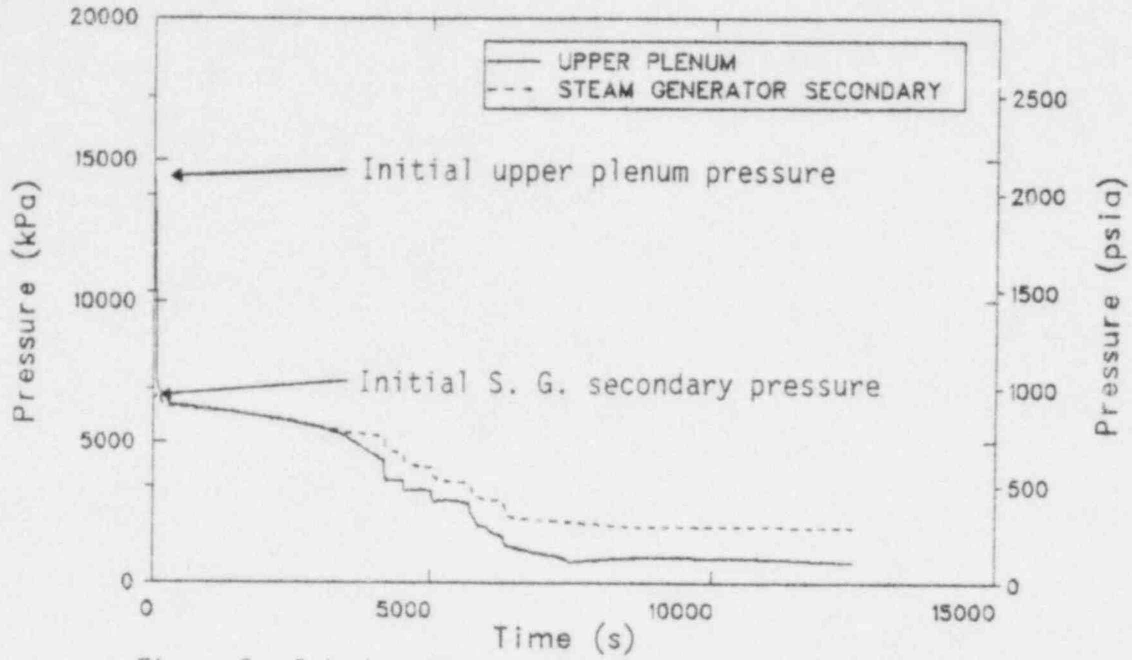

Figure 3. Calculated upper plenum and steam generator pressures -- small break.

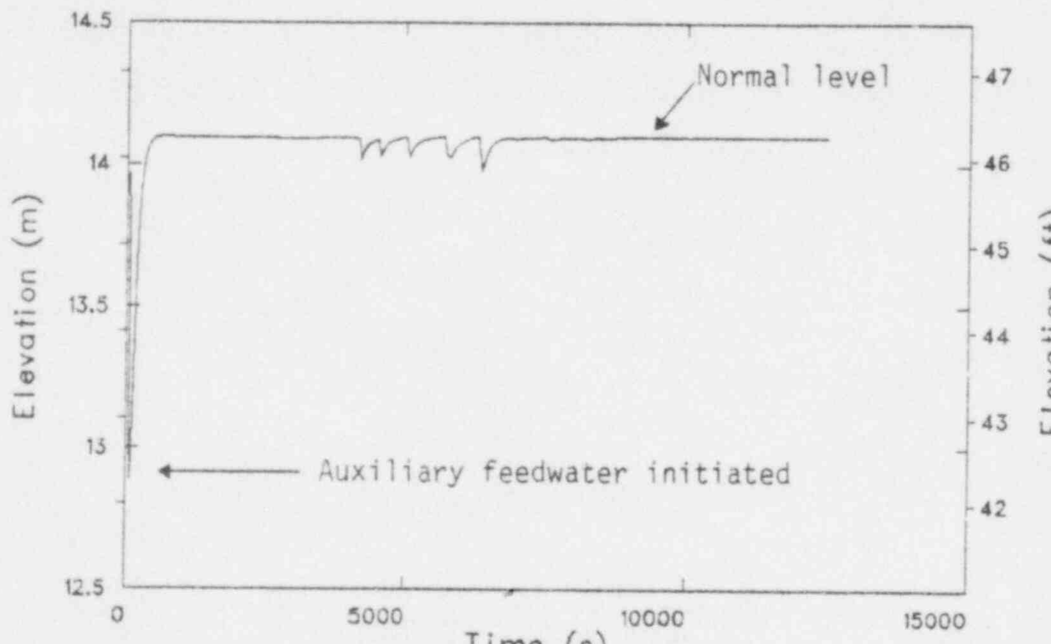

Figure 4. Calculated secondary mixture level -- small break.

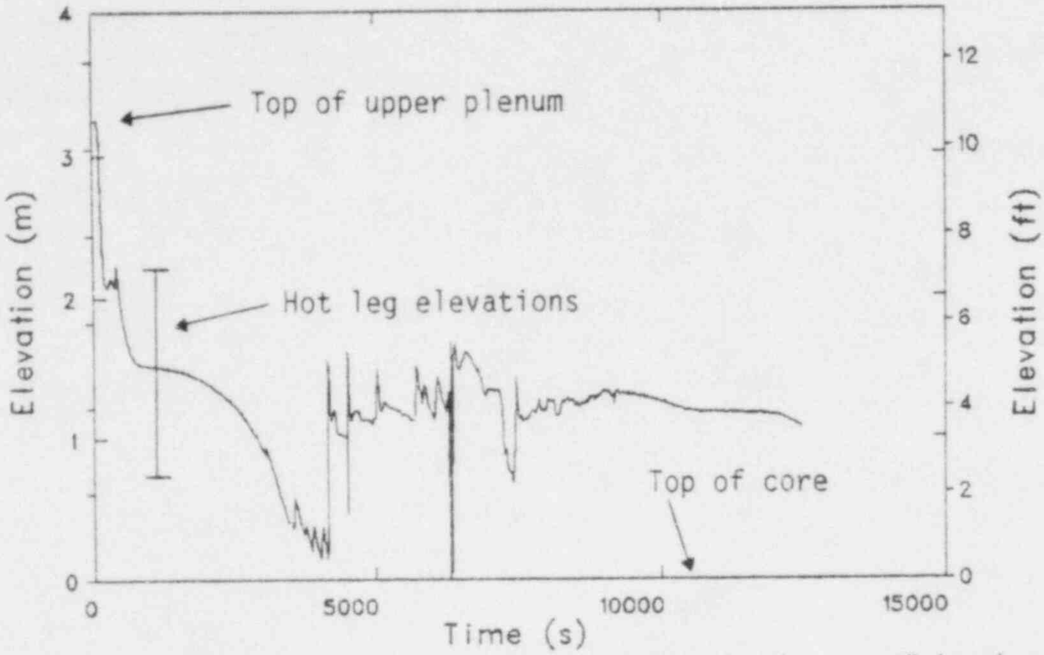

Figure 5. Calculated upper plenum mixture level -- small break.

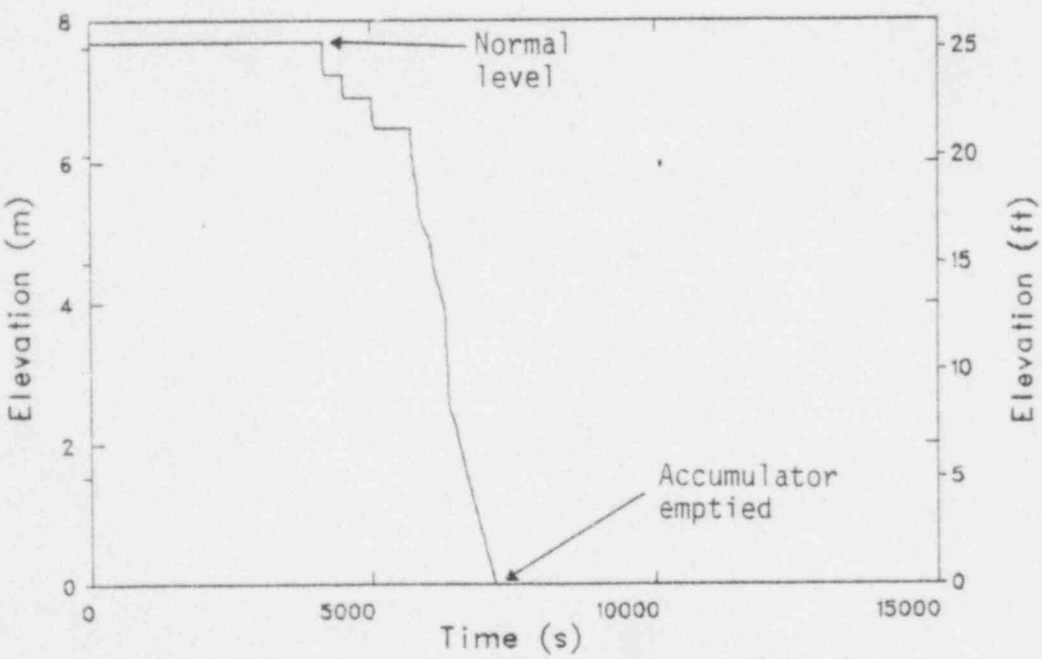

Figure 6. Calculated accumulator level -- small break.

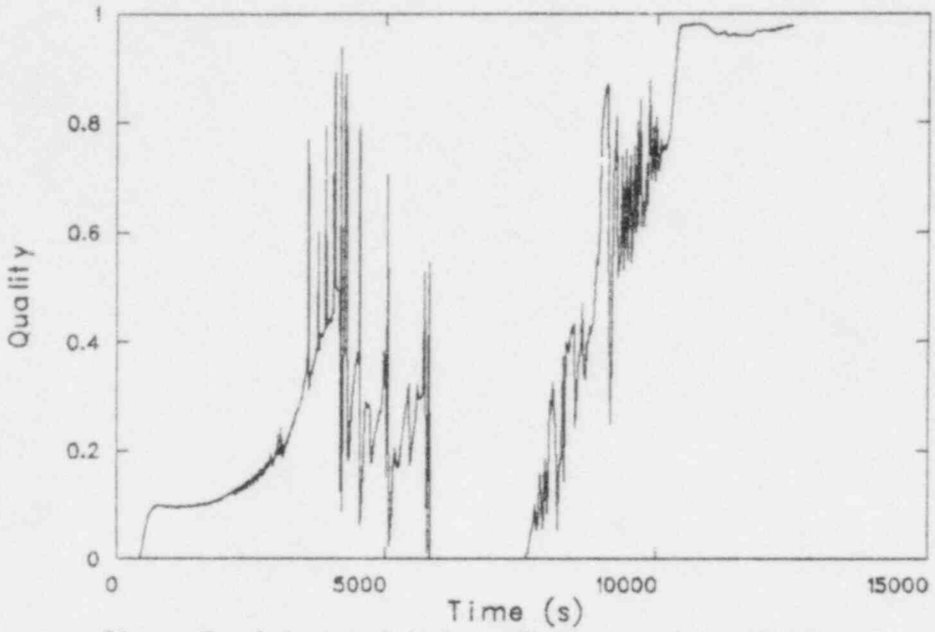

Figure 7. Calculated leak quality -- small break.

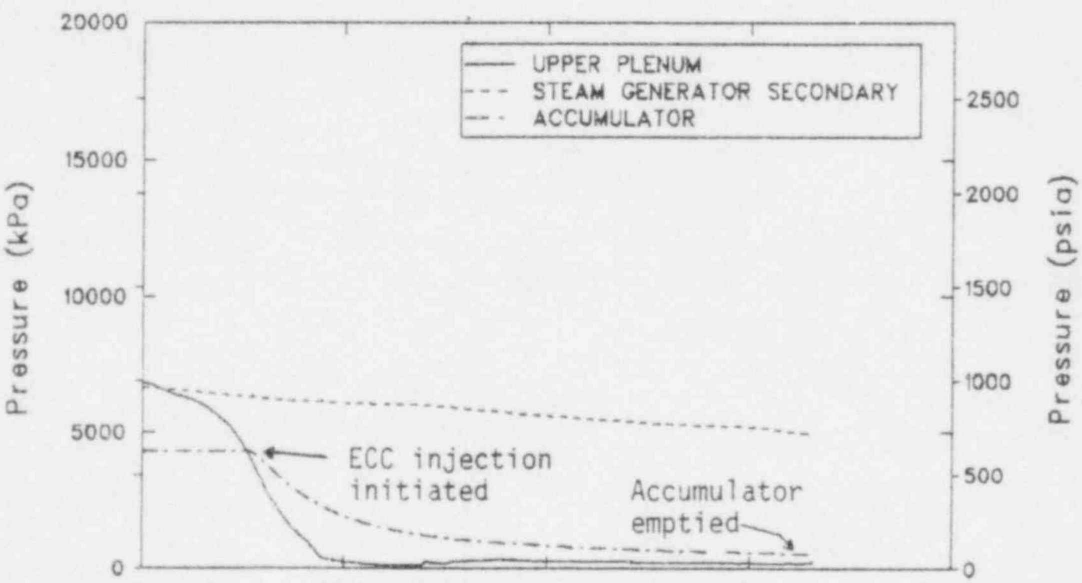

Figure 8. Calculated upper plenum, steam generator, and accumulator pressure -- large break.

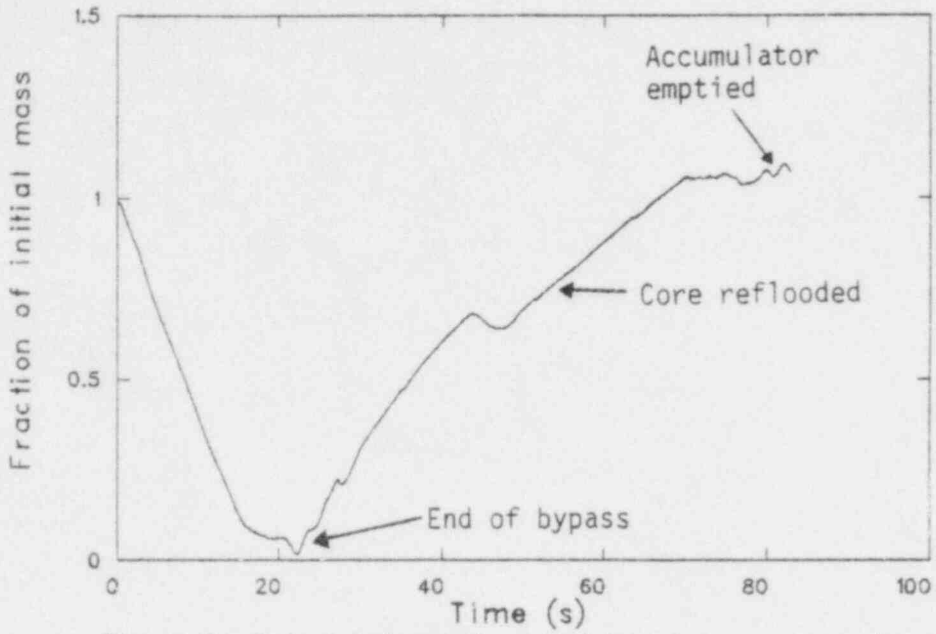

Figure 9. Calculated vessel coolant inventory -- large break.

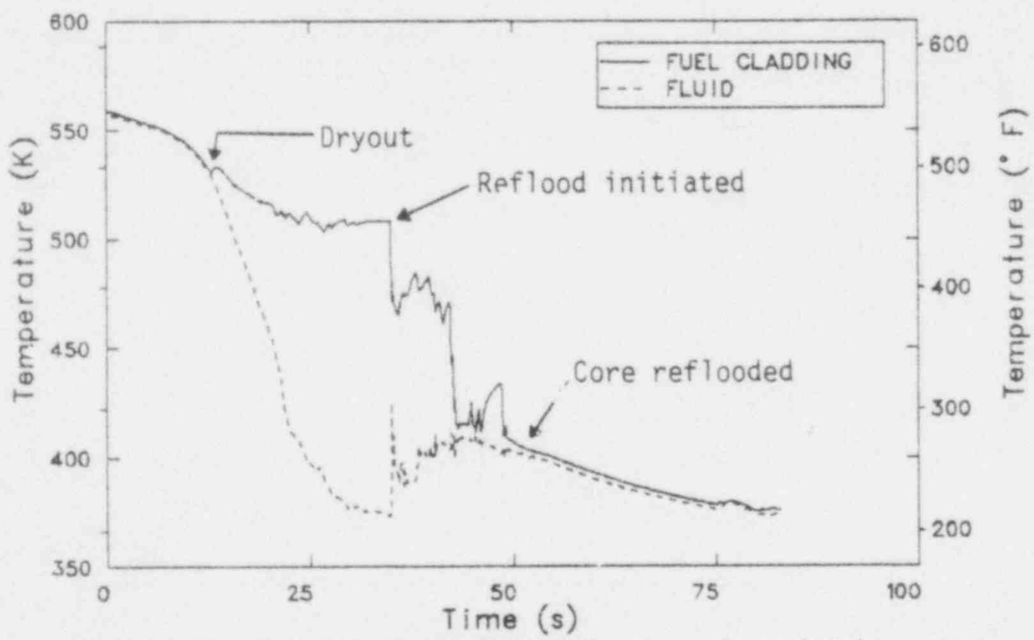

Figure 10. Calculated core temperatures -- large break.

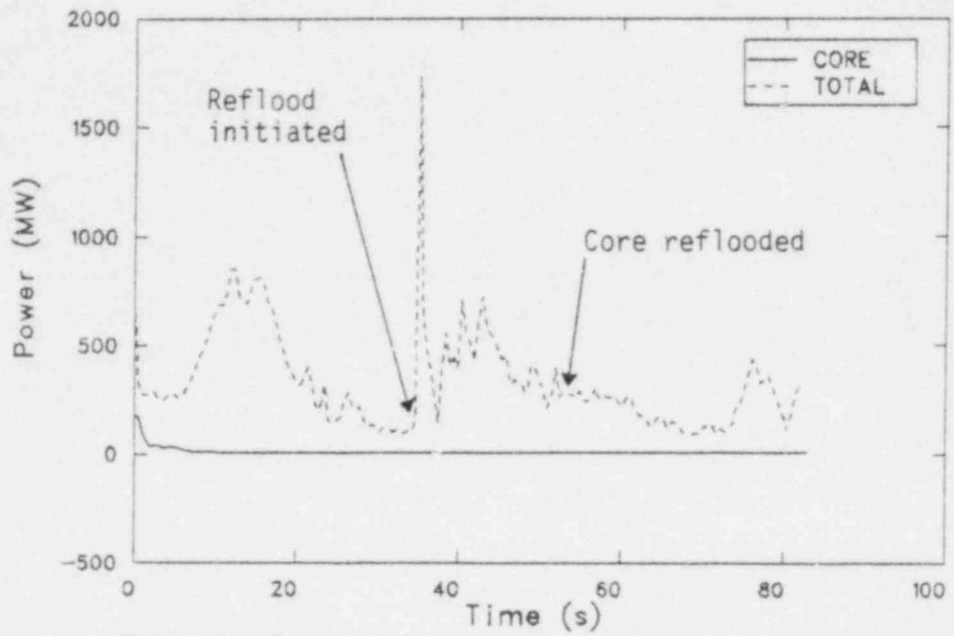

Figure 11. Calculated core and total power -- large break.

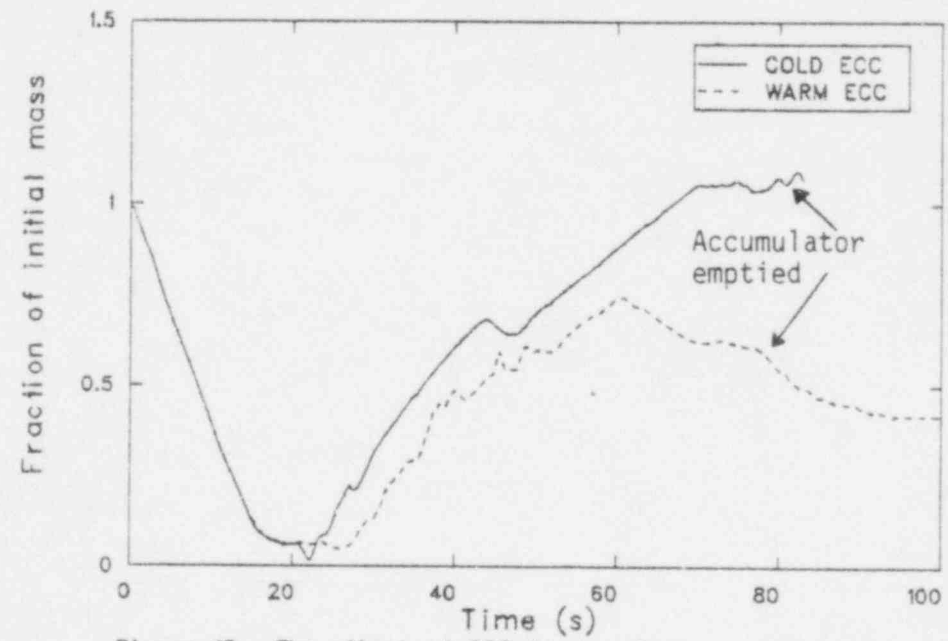

Figure 12. The effect of ECC temperature on vessel coolant inventory -- large break.

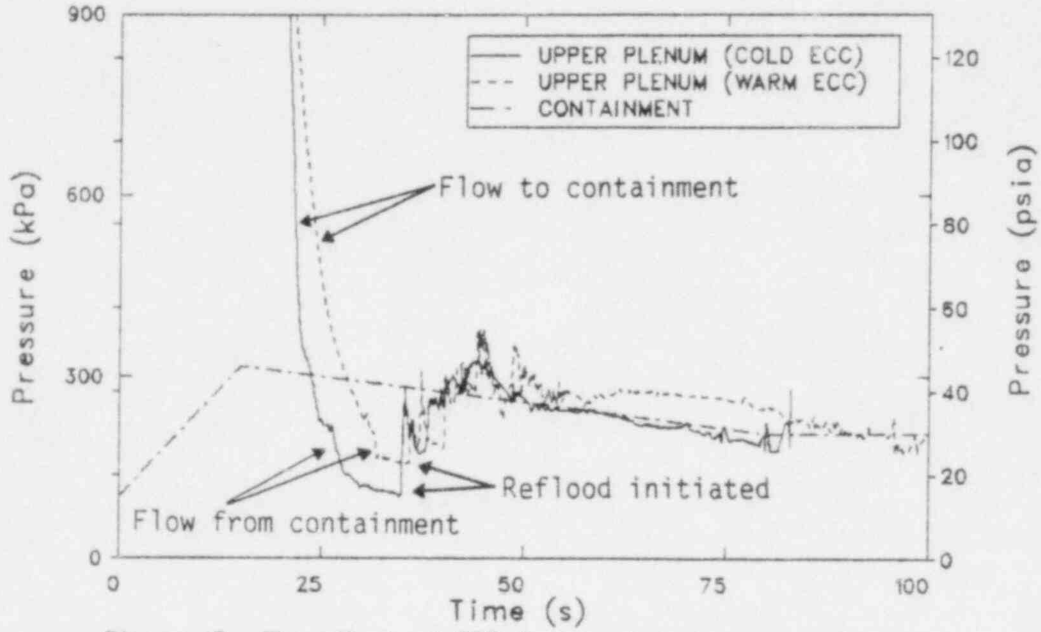

Figure 13. The effect of ECC temperature on upper plenum pressure -- large break.