NUREG/CR-4946
EGG-2510
August 1987

Davis-Besse Uncertainty Study

Cliff B. Davis

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EG&G Idaho

Work performed under
DOE Contract No. DE-AC07-76ID01570
for the **U.S. Nuclear Regulatory Commission**

Idaho National Engineering Laboratory

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NUREG/CR-4946
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Distribution Category: R4

DAVIS-BESSE UNCERTAINTY STUDY

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Published August 1987

EG&G Idaho Inc.
Idaho Falls, Idaho 83415

Prepared for the
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6827

ABSTRACT

The uncertainties of calculations of loss-of-feedwater transients at Davis-Besse Unit 1 were determined to address concerns of the U.S. Nuclear Regulatory Commission relative to the effectiveness of feed and bleed cooling. Davis-Besse Unit 1 is a pressurized water reactor of the raised-loop Babcock & Wilcox design. A detailed, quality-assured RELAP5/MOD2 model of Davis-Besse was developed at the Idaho National Engineering Laboratory. The model was used to perform an analysis of the loss-of-feedwater transient that occurred at Davis-Besse on June 9, 1985. A loss-of-feedwater transient followed by feed and bleed cooling was also calculated. The evaluation of uncertainty was based on the comparisons of calculations and data, comparisons of different calculations of the same transient, sensitivity calculations, and the propagation of the estimated uncertainty in initial and boundary conditions to the final calculated results.

FIN No. A6827—International Code Assessment: RELAP5 and TRAC-BWR

SUMMARY

On June 9, 1985, a loss-of-feedwater (LOFW) transient occurred at Unit 1 of the Davis-Besse Nuclear Power Station. Davis-Besse Unit 1, owned and operated by the Toledo Edison Company, is a pressurized water reactor (PWR) of the raised-loop Babcock & Wilcox (B&W) design with a rated core power of 2772 MWt. This transient, which was initiated from 92% power, resulted in a temporary but total loss of main and auxiliary feedwater. Auxiliary feedwater was eventually restored, and the plant was taken to a safe and stable condition.

Because of the potential severity of the event, the U.S. Nuclear Regulatory Commission (NRC) immediately began a program to analyze the Davis-Besse LOFW transient, including parametric variations. These parametric variations were primarily related to the use of feed and bleed cooling. Feed and bleed cooling, which involves starting the makeup and high-pressure injection (HPI) pumps and opening the pilot-operated relief valve (PORV) located on the top of the pressurizer, would have been used to remove decay heat from the core if auxiliary feedwater had not been restored during the LOFW transient at Davis-Besse. The NRC pursued a two-pronged thermal-hydraulic analysis effort of the LOFW event: an in-house analysis of the event performed through the NRC Office of Nuclear Reactor Regulation (NRR), and an independent analysis performed at Los Alamos National Laboratory (LANL). The NRR analysis utilized the Nuclear Plant Analyzer (NPA) and the RELAP5/MOD2 thermal-hydraulic computer code. The interactive features of the NPA allowed 15 calculations to be completed in a short period of time. The LANL analysis utilized the TRAC-PF1/MOD1 computer code. Both analyses were completed quickly to provide a rapid assessment of the effectiveness of feed and bleed cooling at Davis-Besse. For convenience, the NRR calculations will hereafter be referred to as the R585 calculations because they were performed with RELAP5 and completed in 1985. The calculations performed by LANL will be referred to as the TRAC calculations.

Although the R585 and TRAC calculations indicated that feed and bleed could successfully cool the core if initiated early enough, the NRC realized that there were several uncertainties in the calculations. One source of uncertainty was due to the code input models used to make the calculations. Both the R585 and TRAC calculations were performed with models based on Oconee Unit 1, a lowered-loop B&W PWR, that were quickly modified to represent Davis-Besse. The modifications to the Oconee model that resulted in the R585 model were performed at the Idaho National Engineering Laboratory (INEL). The NRC Office of Nuclear Regulatory Research asked that INEL assess the uncertainties in the R585 calculations. The INEL developed a detailed, quality-assured RELAP5/MOD2 model of Davis-Besse, referred to as the R586 model because it is a RELAP5 model that was developed in 1986. The R586 model was then used to repeat the R585 calculation of the Davis-Besse LOFW transient and a R585 feed and bleed calculation. The feed and bleed calculation represented a LOFW event initiated from 100% power, with feed and bleed started 20 minutes after the beginning of the transient. The calculations repeated by the INEL will hereafter be referred to as the R586 calculations because they were performed with the R586 model.

The R586 calculation of the Davis-Besse LOFW transient was in good qualitative and quantitative agreement with the measured data. The trends observed in the plant were well represented in the calculation. The maximum deviation between calculated and measured reactor coolant system (RCS) pressure was about 0.3 MPa (50 psi). The deviations between calculated and measured RCS temperatures were generally less than 3 K (6°F). Even though different thermal-hydraulic computer codes and input models were used, the R586, R585, and TRAC calculations were similar and showed trends like those observed in the plant. The differences that were observed between the calculations were primarily due to the assumption of different core powers, feedwater flows, and pressurizer spray flows.

The R586 feed and bleed calculation exhibited the phenomena expected in a LOFW event. The secondary side of the once-through steam generators (OTSGs) dried out 130 s after reactor trip, resulting in a heatup of the RCS. After feed and bleed was initiated at 1200 s, the RCS pressurized until the pressurizer safety relief valves (SRVs) opened. The liquid lost through the PORV and SRVs caused the mixture level in the RCS to drop below the pressurizer surge line. The resulting flow of steam through the PORV caused the RCS to depressurize after 3750 s. The makeup pumps began refilling the RCS at 5160 s. The RCS pressure dropped below the shutoff head of the HPI pumps at 5500 s, causing a more rapid refill of the RCS. Feed and bleed successfully cooled the core, which was covered with liquid throughout the calculation.

The R586, R585, and TRAC calculations all indicated that feed and bleed could successfully remove core decay heat if a total LOFW event occurred at Davis-Besse. However, some significant differences between the R586 calculation and the R585 and TRAC

calculations were observed, particularly with respect to event timing and RCS pressure response. These differences affected the course, but not the ultimate outcome, of the transient and were attributed to differences in the boundary conditions. In particular, the core decay power was too small in the R585 calculation after the trip of the reactor coolant pumps because of an error in the input model. The PORV flow was thought to be too small in the TRAC calculation. The comparison of calculations indicated that the specific results were sensitive to the boundary conditions. However, the macroscopic results of all the calculations were similar in that feed and bleed successfully cooled the core.

The uncertainty in the R585 and R586 feed and bleed calculations was evaluated. Several potential sources of uncertainty were identified which could contribute to the overall uncertainty in the feed and bleed calculations. These potential sources of uncertainty included the thermal-hydraulic computer code, the code input model, the initial conditions and boundary conditions of the calculation, the code user, and the assumed transient. The important parameters that determine the thermal-hydraulic signature of a feed and bleed transient were identified. These parameters included the RCS temperature at the initiation of feed and bleed, the ability to depressurize, and the minimum liquid level in the reactor vessel. The uncertainties in the important parameters were estimated based on several factors, including subjective judgments. Better estimates of uncertainty would be obtained from more extensive comparisons of feed and bleed calculations and experimental data.

The uncertainties in the R586 feed and bleed calculation were thought to be relatively small. The uncertainty in the calculated collapsed liquid level in the reactor vessel was estimated to be 1 m (3 ft). The uncertainty in the RCS temperature when feed and bleed was initiated in the R586 calculation was estimated to be 5 K (9°F). This uncertainty corresponds to 11% uncertainty in the calculated RCS heatup rate after OTSG dryout. The corresponding uncertainty in the time required to reach the RCS temperature at which feed and bleed was initiated was about 2 min. These uncertainties were caused by uncertainty in the initial and boundary conditions, primarily the initial OTSG liquid inventory. The uncertainty in the initial and boundary conditions did not alter the results of the R586 calculation relative to the ability to depressurize the RCS during feed and bleed.

The uncertainties in the R585 calculations were estimated to be larger than in the R586 calculation because of the error in core decay power following reactor coolant pump trip as discussed previously. This error caused a bias in the R585 calculations in addition to the uncertainty associated with the initial and boundary conditions. The bias in the RCS temperature at the initiation of feed and bleed ranged from 2 K (4°F) to 11 K (20°F), depending on the time between the trip of the reactor coolant pumps and the initiation of feed and bleed. The corresponding bias in the time required to reach the RCS temperature at which feed and bleed was initiated ranged from 1 to 6 min.

The above estimates of uncertainty are valid for the transients analyzed based on the assumed initiating event, equipment performance, and operator actions. If different assumptions were made regarding these parameters, the differences in the calculated results could exceed the estimated uncertainties.

ACKNOWLEDGMENTS

The author wishes to thank the following people for their contributions to this report: M. Bolander, for the development of the thermal-hydraulic model of the secondary system; C. Dobbe, for the analysis of the loss-of-feedwater transient at Davis-Besse; E. Jenkins, for the generation of the report graphics; M. Waterman, for the development of the model of the integrated control system; and P. Wheatley, for technical support and review. The author also thanks personnel from the Babcock and Wilcox Company and the Toledo Edison Company for their cooperation and support in providing the information used to develop the model of Davis-Besse. The contributions of E. Throm, of the NRC Office of Nuclear Reactor Regulation, and J. Lime, of Los Alamos National Laboratory, are also gratefully acknowledged.

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NOMENCLATURE

AEV	atmospheric exhaust valve
AFW	auxiliary feedwater
ARTS	anticipatory reactor trip system
B&W	Babcock and Wilcox
DADS	data acquisition display system
ECCS	emergency core cooling system
EPRI	Electric Power Research Institute
HPI	high-pressure injection
HPVV	high point vent valve
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LOFW	loss of feedwater
LPI	low-pressure injection
MFP	main feed pump
MSIV	main steam isolation valve
NPA	Nuclear Plant Analyzer
NRC	U.S. Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OTSG	once-through steam generator
PORV	pilot-operated relief valve
PWR	pressurized water reactor
Q_c	volumetric flow due to steam production in the core
Q_m	volumetric flow of makeup
Q_p	volumetric flow out the PORV
RCP	reactor coolant pump
RCS	reactor coolant system
RELAP	Reactor Excursion and Leak Analysis Program
R585	RELAP5 completed in 1985
R586	RELAP5 completed in 1986
SFRCS	steam and feed rupture control system
SRV	safety relief valve
TBV	turbine bypass valve
TRAC	Transient Reactor Analysis Code

DAVIS-BESSE UNCERTAINTY STUDY

1. INTRODUCTION

On June 9, 1985, a loss-of-feedwater (LOFW) event occurred at Unit 1 of the Davis-Besse Nuclear Power Station.[1] Davis-Besse Unit 1, owned and operated by the Toledo Edison Company, is a pressurized water reactor (PWR) of the raised-loop Babcock & Wilcox (B&W) design with a rated core power of 2772 MWt. At the time of the LOFW event, Davis-Besse was operating at 92% power. The transient was initiated by an overspeed of one main feed pump (MFP), which caused the pump to trip, eventually resulting in a reactor trip. Subsequent failures then caused the complete loss of all feedwater. Without feedwater, the reactor coolant began to heat up. Auxiliary feedwater was restored about 20 min after the MFP trip, and the plant was taken to a safe condition.

Because of the potential severity of the event, the U.S. Nuclear Regulatory Commission (NRC) immediately began a program to analyze the Davis-Besse transient, including parametric variations primarily related to the use of feed and bleed cooling. Feed and bleed cooling, which involves opening the pilot-operated relief valve (PORV) located on the top of the pressurizer and starting makeup pumps and high-pressure injection (HPI) pumps, would have been used to remove decay heat from the Davis-Besse core if auxiliary feedwater had not been restored. The NRC pursued a two-pronged thermal-hydraulic analysis effort of the LOFW event at Davis-Besse. First, the NRC performed an in-house analysis of the event through the NRC Office of Nuclear Reactor Regulation (NRR). Second, an independent analysis[2] was performed at the Los Alamos National Laboratory (LANL). Both analyses used thermal-hydraulic computer codes as their primary calculational tool. NRR utilized the Reactor Excursion and Leak Analysis Program (RELAP5/MOD2),[3] while LANL used the Transient Reactor Analysis Code (TRAC-PF1/MOD1).[4] The NRR analysis relied heavily on the Nuclear Plant Analyzer (NPA).[5] The interactive features of the NPA allowed many calculations to be completed in a short period of time following the LOFW transient at Davis-Besse. A simulation of the LOFW event that occurred at Davis-Besse was performed in both analyses.

LANL calculated four parametric variations of the event. These parametric calculations started from 90% rated power, close to the power at the start of the Davis-Besse LOFW transient, and assumed that feedwater was not restored. In one calculation, feed and bleed cooling was not initiated, causing the core to uncover. Feed and bleed cooling was initiated at 15, 20, or 35 min after the event started in the other three calculations. The calculations performed by LANL are referred to as the TRAC calculations in this report.

The NRR analysis consisted of 15 calculations, with initial power of either 90% or 100% of rated core power. Parametric calculations investigated the effect of time to initiation of feed and bleed, the effect of makeup flow prior to feed and bleed, and the effect of PORV flow. The NRR calculations are referred to as the R585 calculations in this report because they were performed with RELAP5 in 1985.

The macroscopic results of the R585 and TRAC calculations were similar. The TRAC calculations indicated that feed and bleed would successfully cool the core if it was initiated within 20 min of the start of the transient. LANL also believed that feed and bleed would successfully cool the core if it was initiated within 35 min of the start of the transient, although this result was not calculated directly but instead was based on an extrapolation. The R585 analysis indicated that feed and bleed could successfully cool the core if initiated within 37 min of the LOFW.

Although the R585 and TRAC calculations indicated that feed and bleed could successfully cool the core if initiated early enough, the NRC realized that there were several uncertainties in the calculations. These uncertainties were related to the plant models used in the calculations, the initial and boundary conditions assumed in the calculations, and the uncertainty in the codes used for the calculations. The uncertainty due to the plant model exists because at the time of the Davis-Besse transient the NRC did not have a model of Davis-Besse Unit 1. The R585 and TRAC calculations were performed with models based on the Oconee Unit 1 that were quickly modified to resemble Davis-Besse. (Oconee Unit 1 is a B&W lowered-loop PWR).

The NRC Office of Nuclear Regulatory Research asked that the Idaho National Engineering Laboratory (INEL) assess the uncertainties in the R585 calculations. The assessment of these uncertainties is the subject of this report. The uncertainties in the plant model were assessed by developing a quality-assured RELAP5/MOD2 model of Davis-Besse based on detailed Davis-Besse plant information. This model was then used to repeat the R585 calculation of the Davis-Besse LOFW transient, as well as a LOFW transient from 100% power with feed and bleed initiated 20 min after the start of the transient. These calculations, called the R586 calculations because they were performed with RELAP5 in 1986, were then compared to the results of the R585 and TRAC calculations to assess the uncertainty in the results. Section 2 of this report provides a description of the RELAP5/MOD2 computer code and the quality-assured model of Davis-Besse. The results of the R586 Davis-Besse LOFW transient and feed and bleed calculations are described in Section 3. Section 3 also compares the results of the R585, TRAC, and R586 calculations and provides an assessment of the uncertainty in the calculations. Conclusions are presented in Section 4. References are provided in Section 5.

2. CODE AND MODEL DESCRIPTION

The RELAP5/MOD2 computer code and the R586 Davis-Besse model are described in Sections 2.1 and 2.2, respectively.

2.1 Code Description

The RELAP5/MOD2 computer code was designed for thermal-hydraulic analysis of transients in PWRs and related experimental systems. RELAP5/MOD2 was developed at the INEL to simulate a wide variety of thermal-hydraulic transients involving steam, water, and noncondensible fluid mixtures. RELAP5/MOD2 has been used to simulate small-break and large-break loss-of-coolant accidents, operational transients such as loss of feedwater, and even severe accidents to the point of fuel damage. The code can be used to simulate the primary and secondary systems of a PWR, including the balance of plant. The version of the code used in this analysis was RELAP5/MOD2 Cycle 36.04. An assessment of the code has been performed.[6] RELAP5/MOD2 is described in detail in Reference 3.

RELAP5/MOD2 contains interactive capabilities which, in conjunction with the NPA, allow an analyst to control and display the results of a calculation as it is running. The analyst can interactively control the same major components in a plant model, such as pumps and valves, which an operator can control in the actual plant. The NPA display of a calculation allows an analyst to comprehensively view the results of an integral model and the interactions between different components and systems. The combined capabilities of RELAP5/MOD2 and the NPA approach the performance of an engineering simulator.

2.2 Model Description

A RELAP5 model of the Davis-Besse Unit 1 PWR was developed at the INEL to perform the R586 LOFW calculations described in this report. The transients calculated included the LOFW event that occurred at Davis-Besse on June 9, 1985, and a total LOFW followed by primary feed and bleed. Section 2.2.1 contains a description of the RELAP5 model. Descriptions of the initial and boundary conditions for the LOFW transients are presented in Sections 2.2.2 and 2.2.3. A description of the quality assurance performed for the model appears in Section 2.2.4

2.2.1 Thermal-Hydraulic and Control System Model. A detailed RELAP5 model of Davis-Besse Unit 1 was developed during this task. The model represents all the major flow paths of the primary and secondary coolant systems. The model of the primary coolant system, also called the reactor coolant system (RCS), represents the two-by-four configuration of the plant, i.e., two loops, each containing one hot leg and two cold legs. The two loops are designated loop A and loop B. The pressurizer is connected to loop A. At Davis-Besse, the loops are sometimes referred to as loop 1 and loop 2, with loop 1 corresponding to loop B. The model of the secondary coolant system includes representations of the condensate and main feedwater systems downstream of the deaerator storage tanks, the auxiliary feedwater (AFW) system, the once-through steam generators (OTSGs), and the main steam lines. Heat structures were used to represent all the major metal components in the plant. Many of the plant control systems were also modeled, including a detailed model of the integrated control system (ICS). In addition to the ICS, the pressurizer pressure and level control systems, anticipatory reactor trip system, and the steam and feed rupture control system (SFRCS) were also modeled. The RELAP5 model of Davis-Besse was designed to allow full interactive control of the major components, such as pumps and valves, which an operator can control in the plant. The model contained 201 volumes, 232 junctions, 188 heat structures, and about 1470 control variables.

The Davis-Besse model was based on the experience gained with the RELAP5 model[7] of the Oconee-1 PWR, which was developed and used extensively in the pressurized thermal shock program. The main differences between the Oconee and Davis-Besse models are due to geometrical and physical differences between the plants and due to improvements to the code and modeling techniques since the development of the Oconee model.

A detailed description of the R586 model of Davis-Besse is presented in Appendix A.

2.2.2 Initial Conditions. Two steady-state initialization calculations were performed with the R586 model of Davis-Besse. The calculations were performed at 92% and 100% of full power. The reactor was operating at 92% power just prior to the LOFW event that occurred on June 9, 1985. The feed and bleed calculations were initialized from the 100% power steady state. The calculated and desired initial conditions for the 92% and 100% power steady states are shown in Tables 1 and 2, respectively. The desired initial conditions for the 92% power case were obtained from plant data taken at 1:34:00 a.m. on June 9, 1985. This time was about 15 s prior to the ICS failure which

Table 1. Comparison of desired and calculated initial conditions at 92% power

Parameter	Desired[a]	RELAP5
Core power, MW	2550	2550
Hot leg pressure, MPa (psia)	14.93 (2166.)[b]	14.96 (2170.)
Hot leg temperature, K (°F)	591.3 (604.7)[b]	591.4 (604.8)
Cold leg temperature, K (°F)	566.9 (560.8)[b]	566.7 (560.4)
Pressurizer level, m (in.)	5.037 (198.3)	5.034 (198.2)
Total reactor coolant flow, kg/s (lbm/s)	18396 (40556)	18111 (39929)
Reactor coolant pump speed, rad/s (rpm)	—[c]	125.7 (1200.)
OTSG outlet pressure, MPa (psia)	6.024 (873.8)[b]	6.043 (876.5)[b]
Total feedwater flow, kg/s (lbm/s)	1306 (2878)	1382. (3046.)
Feedwater temperature, K (°F)	505.8 (450.7)	503.9 (447.4)
Steam superheat, K (°F)	38.7 (69.6)[b]	16.5 (29.7)
OTSG mass (each), kg (lbm)	—[c]	14751 (32520)
OTSG operating level, %	60.95[b]	65.28[b]
OTSG startup level, m (in.)	3.881 (152.8)[b]	4.288 (168.8)[b]

a. Desired value taken from Davis-Besse data acquisition display system at 1:34:00 a.m. June 9, 1985.

b. Parameter represents an average of both loops.

c. Desired value unknown.

Table 2. Comparison of desired and calculated initial conditions at 100% power

Parameter	Desired	RELAP5
Core power, MW	2772	2772
Hot leg pressure, MPa (psia)	14.96 (2170.)	14.96 (2170.)
Hot leg temperature, K (°F)	592.2 (606.3)	592.6 (607.1)
Cold leg temperature, K (°F)	565.2 (557.7)	565.9 (559.0)[a]
Pressurizer level, m (in.)	5.080 (200.0)	5.075 (199.8)
Total reactor coolant flow, kg/s (lbm/s)	18067 (39831)	18143 (39998)
Reactor coolant pump speed, rad/s (rpm)	124.1 (1185.)	125.7 (1200.)
OTSG outlet pressure, MPa (psia)	6.38 (925.)	6.348 (920.7)[a]
Total feedwater flow, kg/s (lbm/s)	1477 (3257)	1536. (3386.)
Feedwater temperature, K (°F)	509.9 (458.2)[a]	511.1 (460.4)
Steam superheat, K (°F)	34.7 (62.5)[a]	12.2 (21.9)[a]
OTSG mass (each), kg (lbm)	17740 (38450)	16696 (36809)[a]
OTSG operating level, %	69[a]	67.4[a]
OTSG startup level, m (in.)	—[b]	4.473 (176.1)[a]

a. Parameter represents an average of both loops.

b. Desired value unknown.

caused main feed pump number 1 (MFP-1) to overspeed and about 55 s prior to the trip of MFP-1. The desired initial conditions for the 100% power steady state were taken from a variety of sources, principally the Updated Safety Analysis Report. However, the parameters relating to OTSG performance, including feedwater mass flow rate, feedwater temperature, and steam superheat, were taken from the Davis-Besse full power acceptance test data.

The calculated initial conditions were generally in excellent agreement with the desired initial conditions. For example, the initial conditions at 92% power were generally within the expected uncertainty of the measurements. Exceptions are in the calculated steam temperature and the feedwater flow rate. The calculated phasic steam temperature at the outlet of the OTSG was in good agreement with the data. However, the code unrealistically calculated that a small amount of liquid was entrained into the steam line. The liquid then evaporated in the steam line, reducing the calculated superheat. Because the calculated equilibrium steam superheat was too low, the calculated feedwater flow was about 5% larger than the actual flow.

2.2.3 Boundary Conditions. For the R586 calculation of the Davis-Besse LOFW event of June 9, 1985, all systems were initially in the automatic mode of control except for MFP-2, which was in the manual mode of control. The transient was initiated by an overspeed of MFP-1. The ICS controlled the plant response prior to reactor trip except for the core power. Core power prior to reactor trip was input based on data. After reactor trip, the core power included contributions from decay heat and fission power. The decay heat was based on the 1979 American Nuclear Society standard,[8] assuming infinite operation, and included actinide decay. The fission power after reactor trip was obtained from a separate-effects reactor kinetics calculation. After reactor trip, feedwater flow to the OTSGs was specified based on data. Similarly, the atmospheric exhaust valves (AEVs) were used to limit OTSG pressure to the measured data. The feedwater and pressure boundary conditions were specified because the operator actions which affected these parameters were not well known. Known manual operations taken during the transient were modeled. Specifically, the pressurizer spray was initiated prior to reactor trip, and letdown was isolated and both makeup pumps were actuated 1 s after reactor trip.

For the R586 feed and bleed calculation, all systems were in the automatic mode of control except as described below. The transient was initiated by the trip of both MFPs. The AFW pumps were assumed to fail to deliver flow to the OTSGs. Letdown was isolated at the start of the transient. No makeup flow was allowed until feed and bleed cooling was initiated. Core decay heat was input as a table representing the 1979 American Nuclear Society standard plus actinide decay. The reactor coolant pumps (RCPs) were tripped, representing a manual action, 1 min after hot leg subcooling decreased to 11 K (20°F). The following actions were taken 20 min after the start of the transient to represent the manual initiation of feed and bleed cooling. The PORV and hot leg high point vent valves (HPVVs) were latched open. The pressurizer heaters were tripped off. Maximum makeup flow was obtained by starting both makeup pumps and locking the makeup valve at its wide-open position. The HPI pumps were started and "piggybacked" to the discharge of the low-pressure injection (LPI) pumps to increase the HPI shutoff head. The shutoff head was 12.7 MPa (1835 psia) in the piggyback configuration, 1.3 MPa (190 psi) higher than in the normal configuration.

The capacities of some of the key systems and valves are presented below to document the values used in the calculations. The sources of these values include information supplied by B&W and Toledo Edison during the development of the model. The pressurizer heaters were modeled to provide a maximum power of 1.329 MW. The pressurizer spray valve was sized to pass 0.012 m^3/s (190 gpm) at normal operating conditions. The PORV was modeled to pass 25.2 kg/s (55.5 lbm/s) of saturated steam at 16.10 MPa (2335 psia) and 47.5 kg/s (104.7 lbm/s) of subcooled liquid at 16.46 MPa (2387 psia) and 613 K (644°F). The resulting PORV area was 9.48×10^{-4} m^2 (0.01020 ft^2), with a single-phase liquid discharge coefficient of 0.82 and a two-phase discharge coefficient of 1.0. Each hot leg HPVV was modeled with an area of 1.830×10^{-5} m^2 (0.000197 ft^2), with single-phase and two-phase discharge coefficients of 0.624. The AEV on each steam line was sized to pass 74.1 kg/s (163.3 lbm/s) of steam at 6.2 MPa (900 psia).

2.2.4 Quality Assurance. Several sources of information were used in the development of the RELAP5/MOD2 model of Davis-Besse. These sources included detailed drawings and blueprints, system descriptions, including the Updated Safety Analysis Report, plant acceptance test data, equipment test data, control system calibration data, and conversations with personnel from B&W and Toledo Edison. B&W and Toledo Edison provided nearly all of the information ultimately incorporated into the model.

The RELAP5 model of Davis-Besse was quality assured in several ways. First, the development of each model component was documented on worksheets that include references to drawings and documents described above. Second, the worksheets were independently checked by an analyst other than the one who developed them. Third, the good agreement between the calculated and desired initial conditions at two different power levels lends confidence to the model. Finally, a calculation of the LOFW transient that occurred at Davis-Besse on June 9, 1985, was performed and compared with plant data. This comparison, which appears in Section 3 of this report, provides additional confidence in the model.

3. RESULTS

Results of the R586 calculation of the Davis-Besse LOFW transient are presented in Section 3.1. The R586, R585, and TRAC calculations of the Davis-Besse LOFW transient are compared in Section 3.2. The R586 feed and bleed calculation from 100% power is described in Section 3.3. The R586, R585, and TRAC feed and bleed calculations are compared in Section 3.4. The uncertainty in the feed and bleed calculations is addressed in Section 3.5. Computer run time statistics for the R586 calculations are presented in Appendix B.

3.1 R586 LOFW Transient Calculation

The R586 calculation of the LOFW transient that occurred at Davis-Besse on June 9, 1985, was performed with the RELAP5/MOD2 model described in Section 2. The calculation of this transient is summarized in Table 3, which presents a sequence of key events. Figures comparing the response of the calculation and the plant follow. The plant data shown in the figures were based on computer printouts of the Davis-Besse data acquisition display system (DADS) as digitized by LANL. Time zero in the calculation and the figures corresponds to 1:34:00 a.m. Other sources of data include the plant process computer alarm printouts. It is believed that the clock times between the DADS data and the alarm printouts varied slightly, with the alarms occurring at a wall clock time 6 s later than the corresponding events were indicated by the DADS.

The plant and calculation were steady for 15 s (until 1:34:15 a.m.), when MFP-1 began to overspeed. The exact cause of the overspeed and the subsequent response of MFP-1 are not known, although MFP-1 tripped at 55 s. The MFP-1 overspeed was modeled by linearly increasing MFP-1 speed from its initial value to its overspeed trip setpoint between 15 and 55 s. The effects of the overspeed on measured and calculated feedwater flow are shown in Figures 1 and 2. The overspeed of MFP-1 caused an increase in feed flow to both OTSGs. The ICS responded to the increase in feed flow by partially closing the main feedwater control valves (see Figures 3 and 4), thus reducing the flow. The MFP-1 turbine tripped on overspeed at 55 s, causing an immediate reduction in the flow to both OTSGs. The ICS responded by opening the main feedwater control valves, which caused a slight increase in feedwater flow. The reactor tripped on high RCS pressure at 75 s in the calculation, 9 s earlier than in the plant. The ICS responded to the reactor trip by closing the main feedwater control valves, sharply reducing the feedwater flow. Figures 1 through 4 show that the RELAP5/MOD2 model of the ICS provided an excellent representation of the measured response during the early portion of the transient.

Table 3. Sequence of events for the Davis-Besse LOFW transient calculation

Time (s)	Event
0.0	Calculation starts at 1:34:00 a.m.
15	MFP-1 overspeed begins
55	MFP-1 tripped
71	Pressurizer spray on
75	Reactor tripped
76	MSIVs begin to close
76	Letdown isolated; Second makeup pump started
82	MSIVs fully closed
86	Pressurizer heaters on
89	Spray off
380	Feedwater terminated
460	Pressurizer heaters off
480	Spray on
650	OTSGs dry out
755	PORV opened
765	Calculation terminated

The main steam isolation valves (MSIVs) inadvertently closed shortly after the reactor trip. Thus, the only source of steam available to drive the MFP-2 turbine was the steam stored in the steam line and connecting piping. Because the MFP turbines were not modeled, the turbine response during periods of degraded steam flow was not known, and the operator actions taken to regulate MFP-2 speed were not well characterized, the feedwater flow after reactor trip was not calculated but was input based on data. The feedwater flows used in the calculation were based on the measurements shown in Figures 1 and 2. Zero flow was provided to both OTSGs after 380 s in the calculation, even though the flow measurements did not read zero. The flow was known to be zero after 430 s, indicating a bias in the measurements. An operator

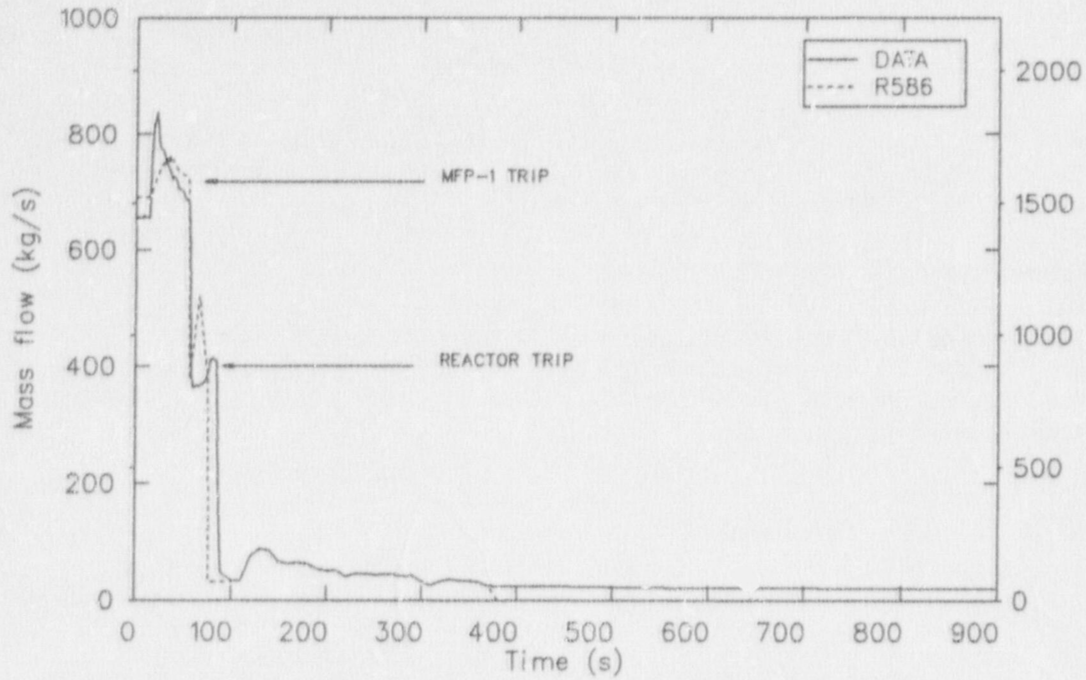

Figure 1. Feedwater flow to OTSG A in the R586 calculation of the Davis-Besse LOFW transient.

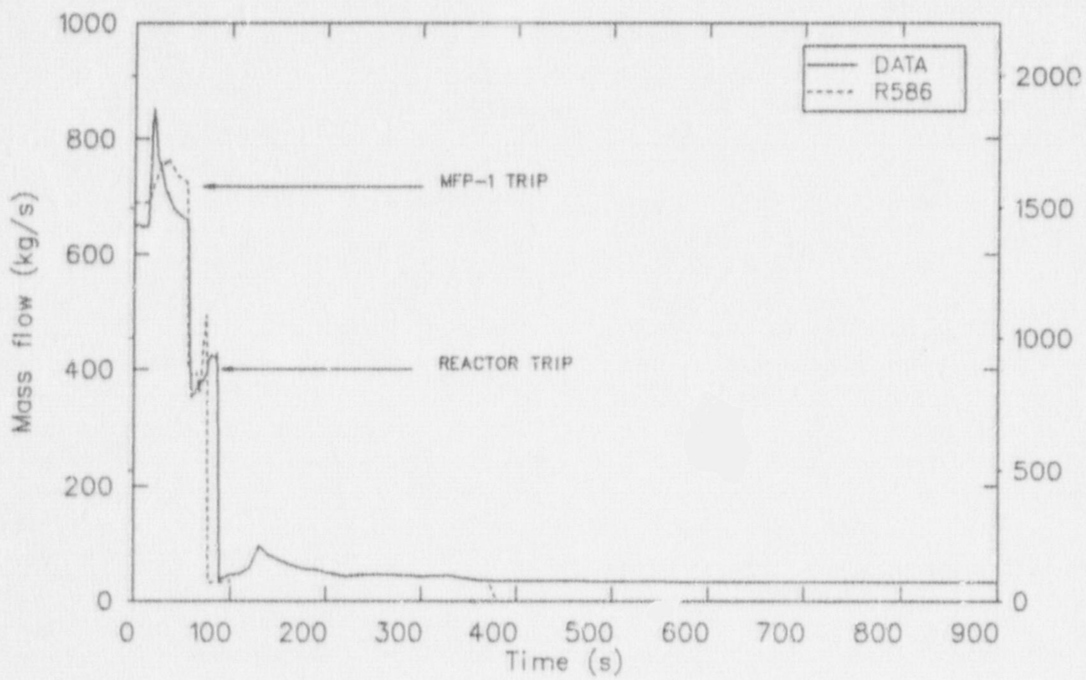

Figure 2. Feedwater flow to OTSG B in the R586 calculation of the Davis-Besse LOFW transient.

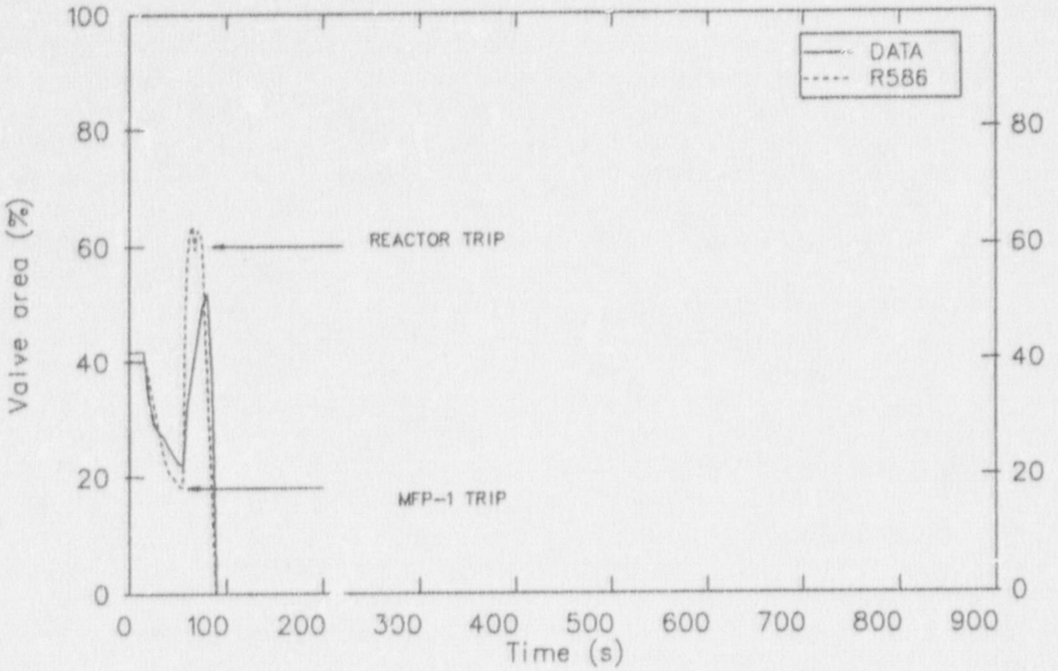

Figure 3. OTSG A main feedwater control valve area in the R586 calculation of the Davis-Besse LOFW transient.

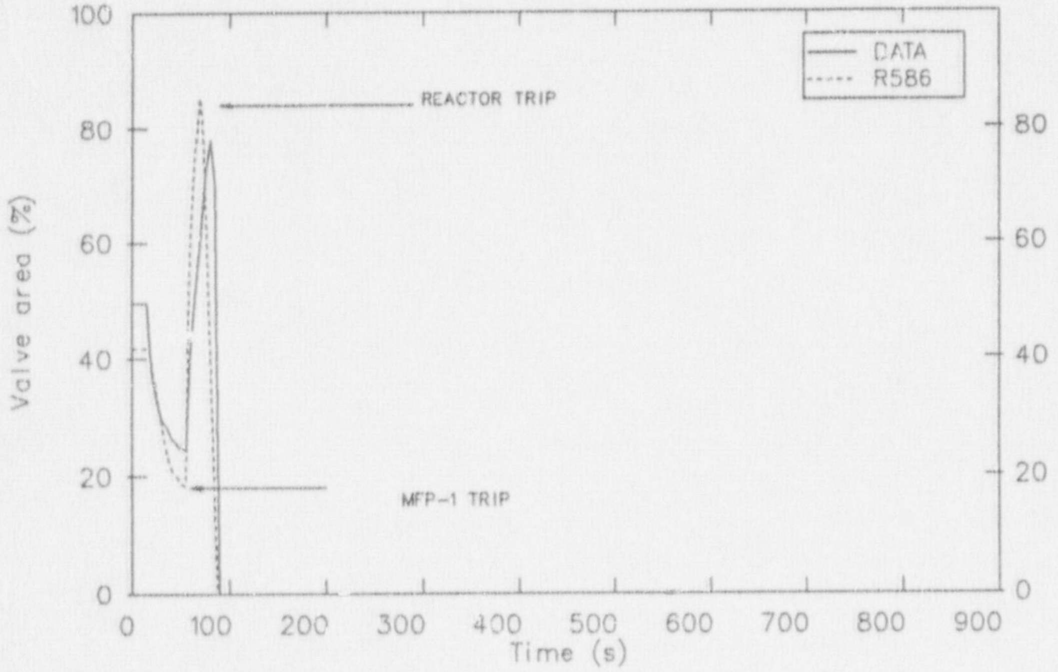

Figure 4. OTSG B main feedwater control valve area in the R586 calculation of the Davis-Besse LOFW transient.

inadvertently actuated SFRCS on low steam pressure at 430 s. SFRCS, responding as if a steam line break had occurred, isolated both OTSGs, terminating feedwater flow. The exact time that feedwater flow decreased to zero in the plant is not known because of the possibility that there was insufficient steam to drive the turbine. However, 380 s appears a reasonable estimate based on the measured flow response and the known bias.

Calculated and measured startup liquid levels are shown in Figures 5 and 6 for the A and B OTSGs, respectively. The calculated and measured levels were generally in excellent agreement. Both calculated and measured levels increased slightly after 15 s and then decreased rapidly after 55 s due to the overspeed and trip of MFP-1. The levels continued to decrease after the reactor trip until a nearly constant value was reached about 120 s. This constant corresponded to the normal posttrip level of 0.89 m (35 in.). The calculated levels were generally too high after 300 s, indicating that too much feed flow was assumed. The levels began to decrease at 380 s when the feedwater flow was terminated. The OTSGs were completely dry at about 650 s in both the calculation and the transient. Note that although the OTSGs were dry, the calculated and measured liquid levels did not reach zero because of the weight of the steam between the differential pressure taps used in the startup level measurement.

The calculated and measured pressures of the OTSGs are illustrated in Figures 7 and 8. A slight pressurization was calculated and observed after the overspeed of MFP-1, followed by a slight depressurization after the MFP-1 trip at 55 s. The reactor trip and turbine trip at 75 s in the calculation resulted in the immediate closure of the turbine stop valves. Within 1 s, the MSIVs also began to close. The closure of the turbine stop valves and the MSIVs resulted in the rapid pressurization of the OTSGs and the opening of the AEVs and the safety relief valves (SRVs). With the reactor tripped and the condenser unavailable, the ICS would normally try to control OTSG pressure at 7.1 MPa (1030 psia). However, the measured pressure varied significantly from the setpoint, either because of operator actions which are not well quantified or anomalous system behavior. Consequently, the OTSG pressure was controlled as a boundary condition after reactor trip in the calculation. The AEVs were opened if the calculated pressure exceeded the measured pressure and closed otherwise. The calculated pressures diverged from the data near the time of OTSG dryout. In the calculation, the pressure increased slowly due to heat transfer to superheated steam. The more rapid pressure increases observed in the data indicated that a small amount of liquid was left in the OTSGs. This liquid then boiled, pressurizing the OTSGs.

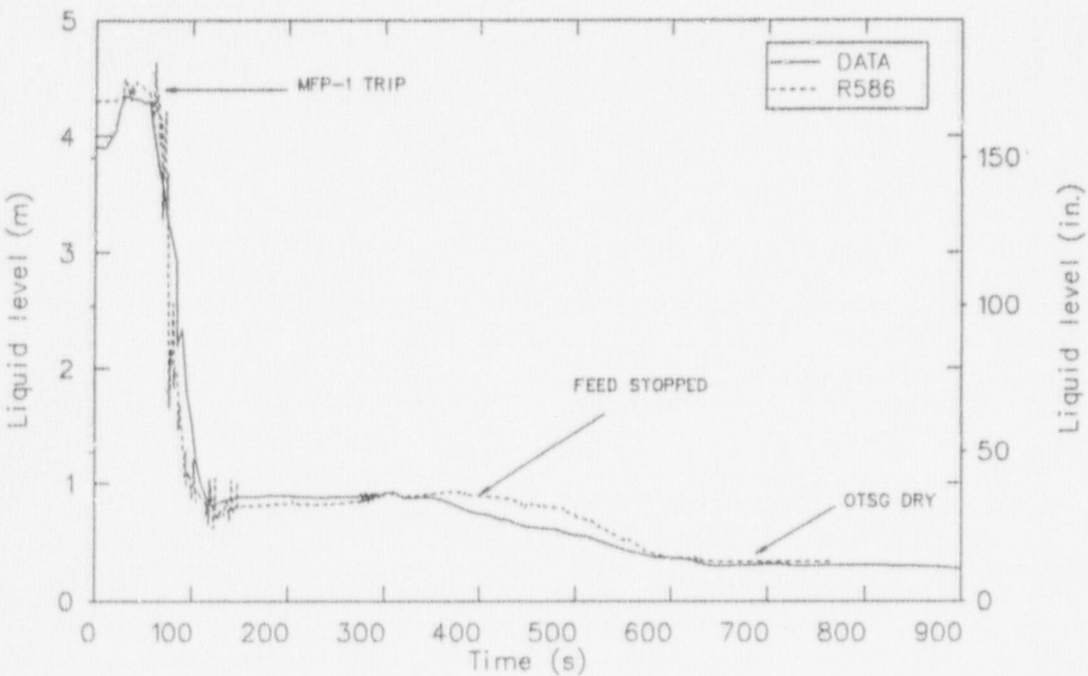

Figure 5. Startup liquid level in OTSG A in the R586 calculation of the Davis-Besse LOFW transient.

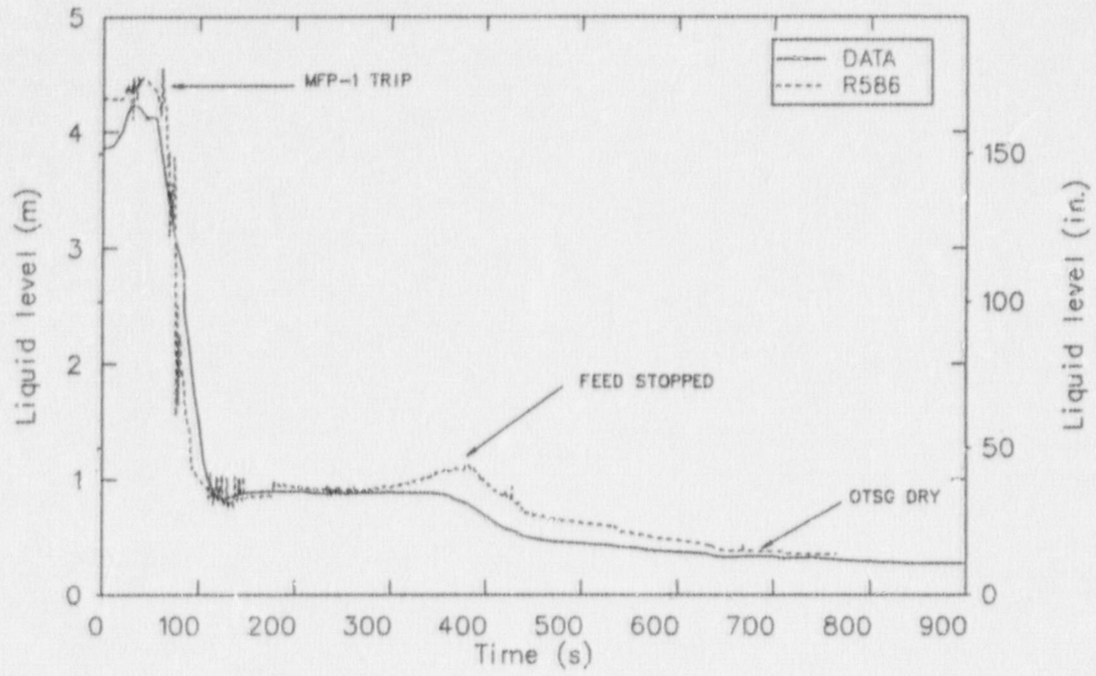

Figure 6. Startup liquid level in OTSG B in the R586 calculation of the Davis-Besse LOFW transient.

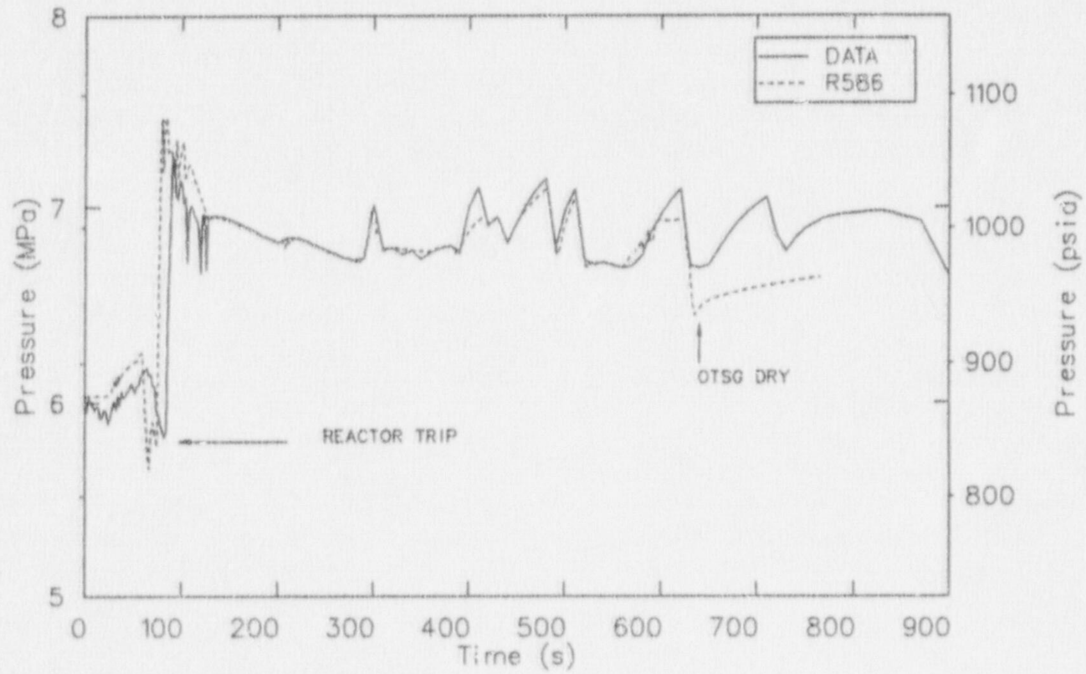

Figure 7. Pressure in OTSG A in the R586 calculation of the Davis-Besse LOFW transient.

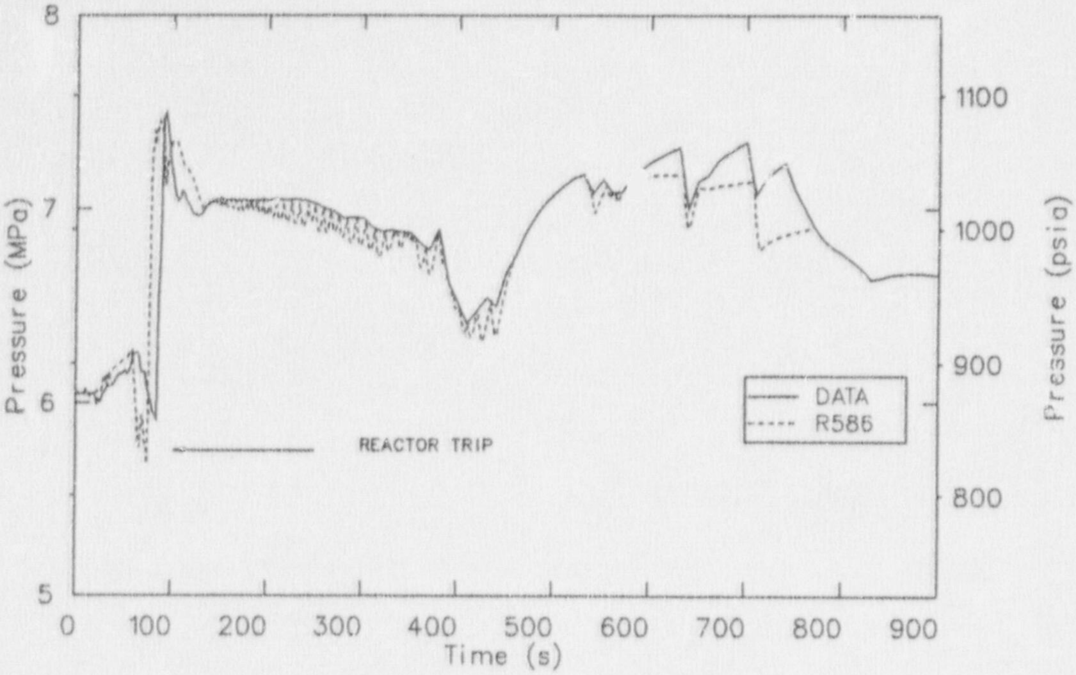

Figure 8. Pressure in OTSG B in the R586 calculation of the Davis-Besse LOFW transient.

Calculated and measured core powers are shown in Figure 9. The measured core power was input as a boundary condition until the time of reactor trip. It was necessary to represent core power as a boundary condition because a reactor kinetics model of Davis-Besse has not yet been developed. Consequently, the model could not represent mechanistically the effects of the MFP-1 overspeed on core power through the influence of control rod movements or moderator temperature feedback. The reactor tripped at 75 s in the calculation on high RCS pressure, slightly before the observed trip at 84 s. The power then decreased rapidly. The measurement was lower than the calculation after 100 s because the measured power was based on the detection of fissions and did not include decay heat. The measured fission power was significant compared to decay heat for approximately 100 s after the reactor trip. The posttrip fission power was associated with the emission of delayed neutrons.

Calculated and measured reactor coolant pressures are shown in Figure 10. The MFP-1 overspeed at 15 s caused a slight depressurization of the RCS. The depressurization was caused by the increase in feedwater flow, which cooled the RCS (see Figures 11 and 12) and reduced the pressurizer liquid level (Figure 13). The trip of MFP-1 at 55 s reduced the feedwater flow, causing a rapid increase in reactor coolant temperature, pressurizer level, and reactor coolant pressure. Pressurizer spray was initiated but was unable to stop the pressure increase. The reactor coolant pressure reached the high pressure setpoint of 15.9 MPa (2300 psia) and the reactor tripped in both the calculation and transient. Shortly after reactor trip, the OTSGs removed more heat from the RCS than was produced in the core. Consequently, the RCS temperature and pressure decreased rapidly, along with the pressurizer level. The operators acted to maintain pressurizer level by isolating letdown and starting the second makeup pump. As the RCS pressure fell, pressurizer spray was automatically terminated and the heaters were actuated. These operator and automatic actions stabilized RCS pressure and level, but the RCS temperature continued to decrease. By 300 s, the pressure, temperature, and level were all lower in the calculation than in the transient. A small reduction in the feedwater flow between 200 and 300 s would improve the calculation of these parameters. The termination of feedwater flow at 380 s in the calculation resulted in an increase in the RCS temperature. The temperatures then increased for the remainder of the calculation. The rate of temperature increase was similar to the data. The calculated and measured rates of increase in pressurizer level were also similar. Pressurizer spray came on near 480 s, briefly halting the pressurization in both the calculation and data. However, the pressure then increased until limited by the PORV. The PORV first opened at 755 s in the calculation and at 880 s in the transient. It was discovered through sensitivity calculations that if the OTSGs were dry, spray initiation could reduce the rate

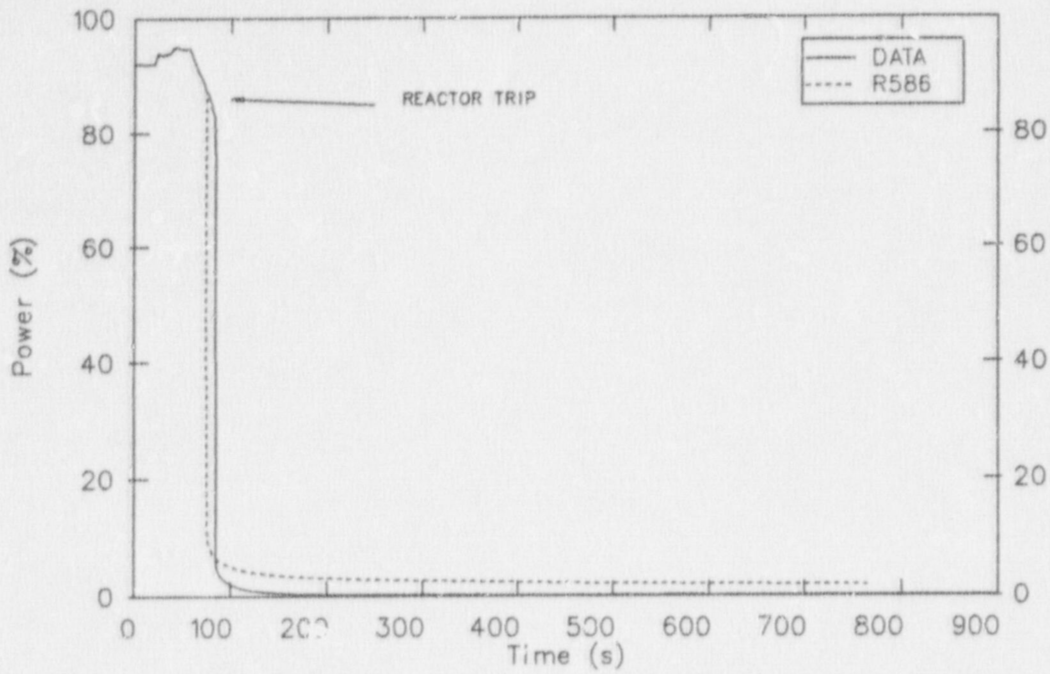

Figure 9. Reactor power in the R586 calculation of the Davis-Besse LOFW transient.

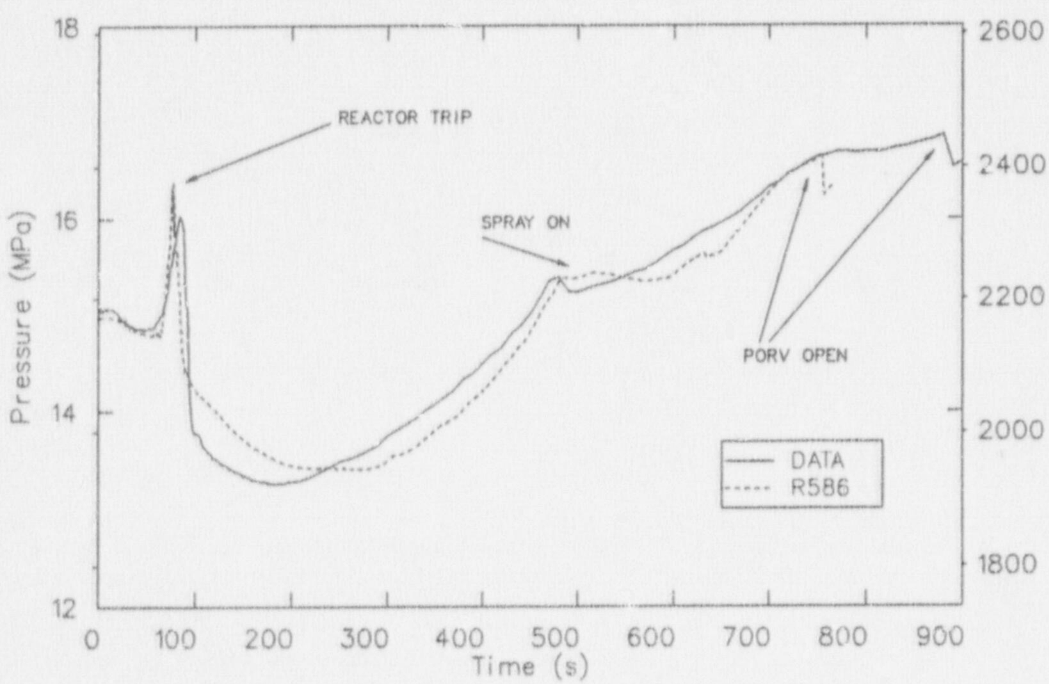

Figure 10. Reactor coolant pressure in the R586 calculation of the Davis-Besse LOFW transient.

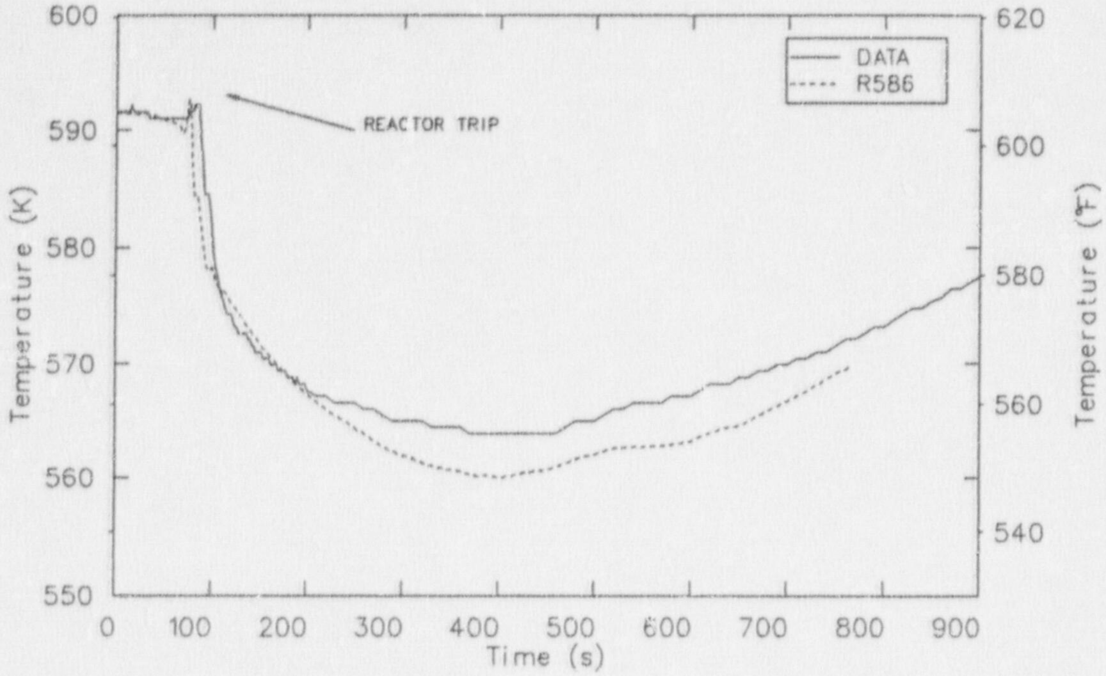

Figure 11. Coolant temperatures in the B loop hot leg in the R586 calculation of the Davis-Besse LOFW transient.

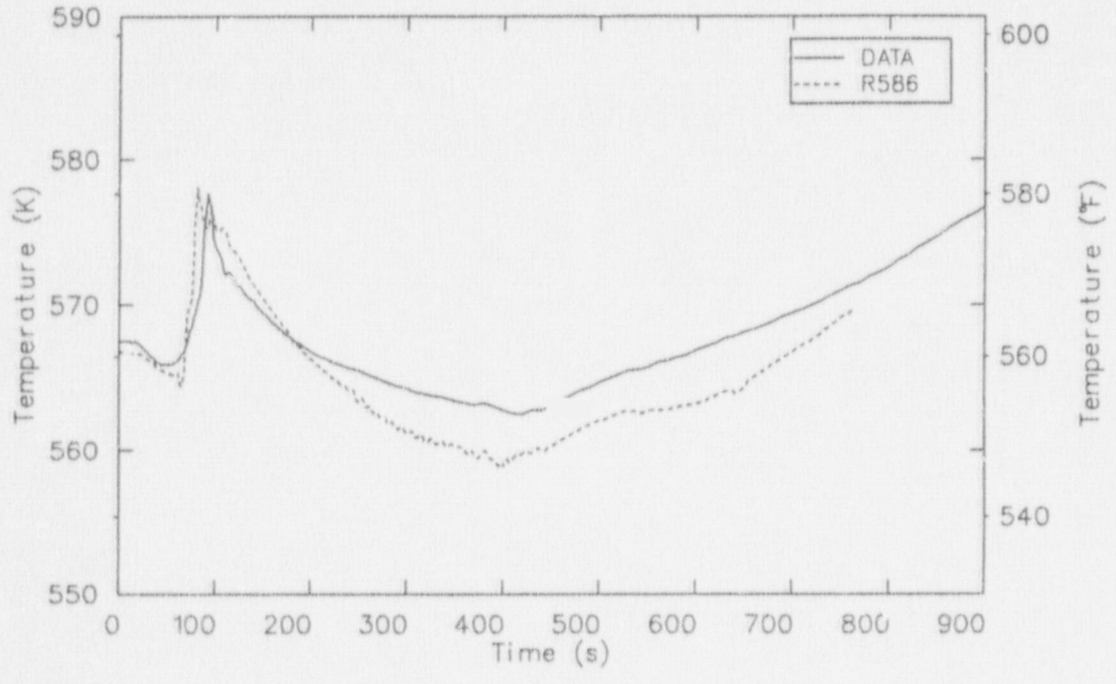

Figure 12. Coolant temperatures in the B loop cold leg in the R586 calculation of the Davis-Besse LOFW transient.

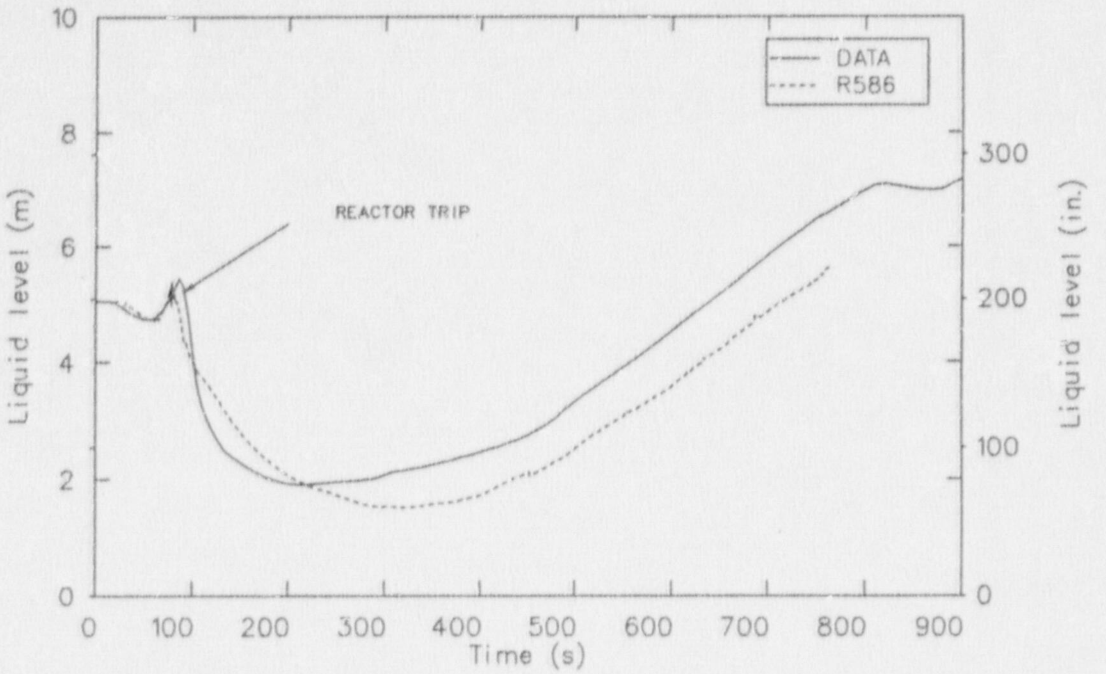

Figure 13. Pressurizer liquid levels in the R586 calculation of the Davis-Besse LOFW transient.

of pressurization but could not depressurize the RCS, as shown in Figure 10.

Calculated and measured hot leg mass flows were in excellent agreement, as illustrated in Figure 14. With the RCPs on, the variations in mass flow were caused by the changes in coolant density associated with the variations in RCS temperature.

A review of Figures 1 through 14 indicates that the calculated trends were in excellent agreement with the data. The magnitudes of the calculated values were also generally in good agreement with the data. For example, the maximum deviation between calculated and measured RCS pressure was about 0.3 MPa (50 psi). The deviation between calculated and measured temperatures was generally less than 3 K (6°F), not significantly larger than the estimated uncertainty in the data. The calculated results were sensitive to the feedwater flow. It would be possible to improve most of the calculated results by adjusting the feedwater flow. In particular, a 30% reduction in the feedwater flow after 200 s would improve the calculated response. However, this improvement would probably not be meaningful, considering the uncertainty in the feed flow and the other data.

3.2 Comparison of Davis-Besse LOFW Transient Calculations

The three calculations (R585, R586, and TRAC) for the LOFW transient that occurred at Davis-Besse on June 9, 1985, are compared in this section. The R586 calculation was discussed in the previous section and represented the transient from 1:34:00 a.m. The TRAC and R585 calculations started 50 to 55 s later, at about the time MFP-1 tripped. Different thermal-hydraulic computer codes and input models were used in the three calculations. The R585 and R586 calculations were performed with the same computer code, RELAP5/MOD2, but with different input models. The TRAC calculation was actually performed with TRAC-PF1/MOD1.

Calculated and measured RCS pressures are compared in Figure 15. The R586 and R585 calculations were generally similar, although two differences were noted. First, the pressure decreased further after reactor trip in the R585 calculation than in the R586 calculation. This difference was believed to be due to the core power boundary condition and will be discussed later. Second, the RCS pressure did not increase to the PORV setpoint in the R585 calculation, but instead was limited by the operation of pressurizer spray, which was significantly larger than in the R586 calculation. The spray valve passed 0.012 m^3/s (190 gpm) in the R586 calculation and about 0.032 m^3/s (500 gpm) in the R585 calculation. The lower value is appropriate for Davis-Besse. The TRAC calculation reached a higher peak pressure prior to reactor trip, then depressurized more slowly than the R586 calculation and the data. According to Reference 2, the higher peak

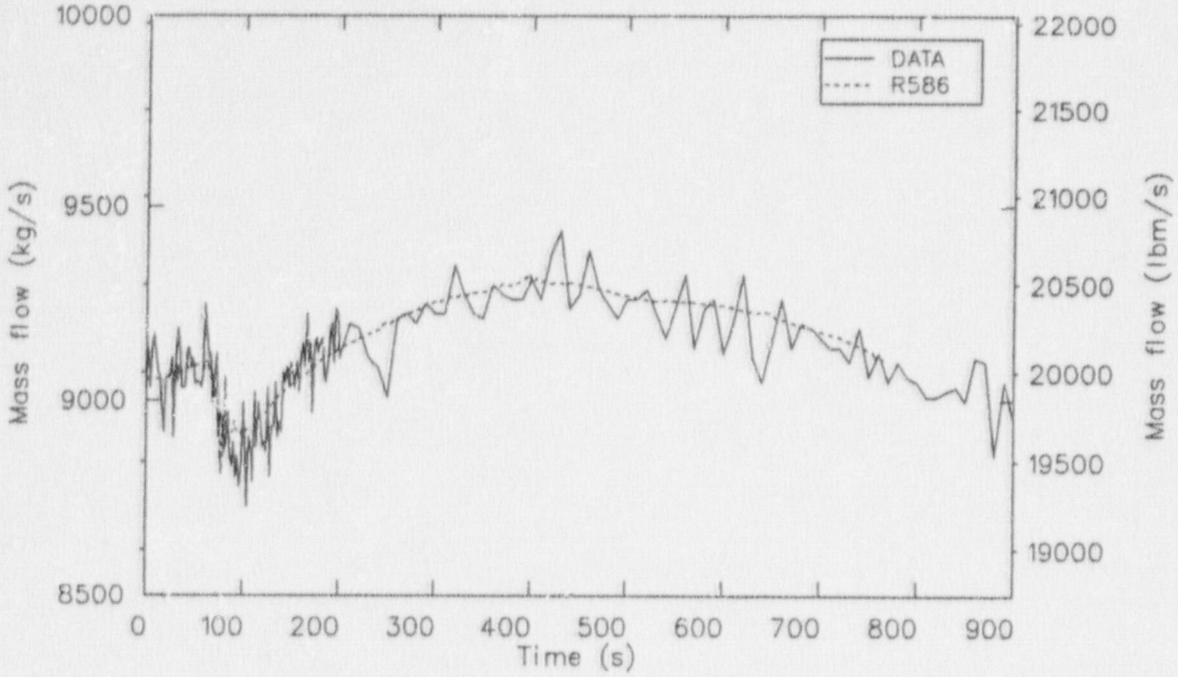

Figure 14. Mass flow in the A loop hot leg in the R586 calculation of the Davis-Besse LOFW transient.

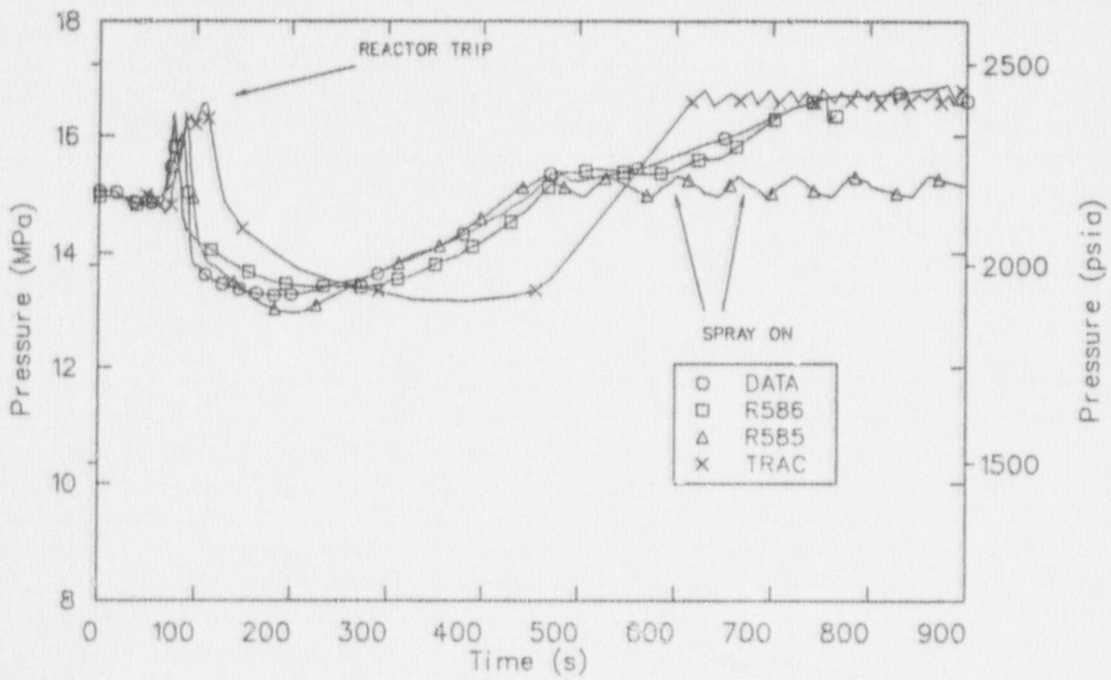

Figure 15. A comparison of calculated reactor coolant pressures during the Davis-Besse LOFW transient.

pressure was caused by the specification of reactor trip on time rather than RCS pressure, which resulted in the addition of too much core power to the RCS. Also, the TRAC pressure increased more rapidly than the data or other calculations after the OTSGs dried out near 500 s. The pressurizer spray was inhibited in the TRAC calculation, and thus the slope of the pressure curve did not change when the open setpoint of the spray valve was reached.

Calculated and measured pressurizer liquid levels are compared in Figure 16. The R586 and R585 calculations were similar except that the level decreased more rapidly after reactor trip in the R585 calculation. A similar difference was observed in the RCS pressure comparison discussed previously. The initial level was significantly too high in the TRAC calculation, but the trends were similar to those discussed for the RCS pressure. The rates of level increase after the OTSGs dried out were similar in all three calculations and the data.

The calculated RCS thermal responses are compared in Figure 17, which shows B loop hot leg temperatures. The R586 and R585 calculations were again quite similar except that the temperature decreased more rapidly after reactor trip in the R585 calculation. The TRAC calculation decreased slower after reactor trip. The rates of temperature increase after OTSG dryout were similar. The deviations in the rates of temperature decrease after reactor trip were caused by variations in the OTSG liquid levels, as shown in Figure 18. The level in the TRAC calculation was significantly lower after reactor trip than in the other calculations and the data. Part of this discrepancy is due to the fact that the parameters shown are not directly comparable. The TRAC curve represents collapsed liquid level, while the R586 and R585 curves mimic the output of the plant instrumentation by including the head of steam between the differential pressure taps as part of the indicated level. However, a comparison of collapsed liquid levels in the R586 and TRAC calculations at 300 s revealed that the level was significantly lower in the TRAC calculation. The lower OTSG level contributed to slower rates of decrease in RCS pressure, temperature, and pressurizer level.

The results shown in Figure 17 showed that the calculated results are sensitive to OTSG level, which in turn depends on feedwater flow. However, the variations in OTSG level between the R586 and TRAC calculations were surprising, considering that the initial OTSG liquid inventories were nearly identical in both calculations and the input feedwater flows were based on the measured flow rates. The discrepancy in OTSG levels may have been a result of the excess core power in the TRAC calculation near the time of reactor trip. The OTSG level in the R585 calculation was generally lower than in the R586 calculation. This

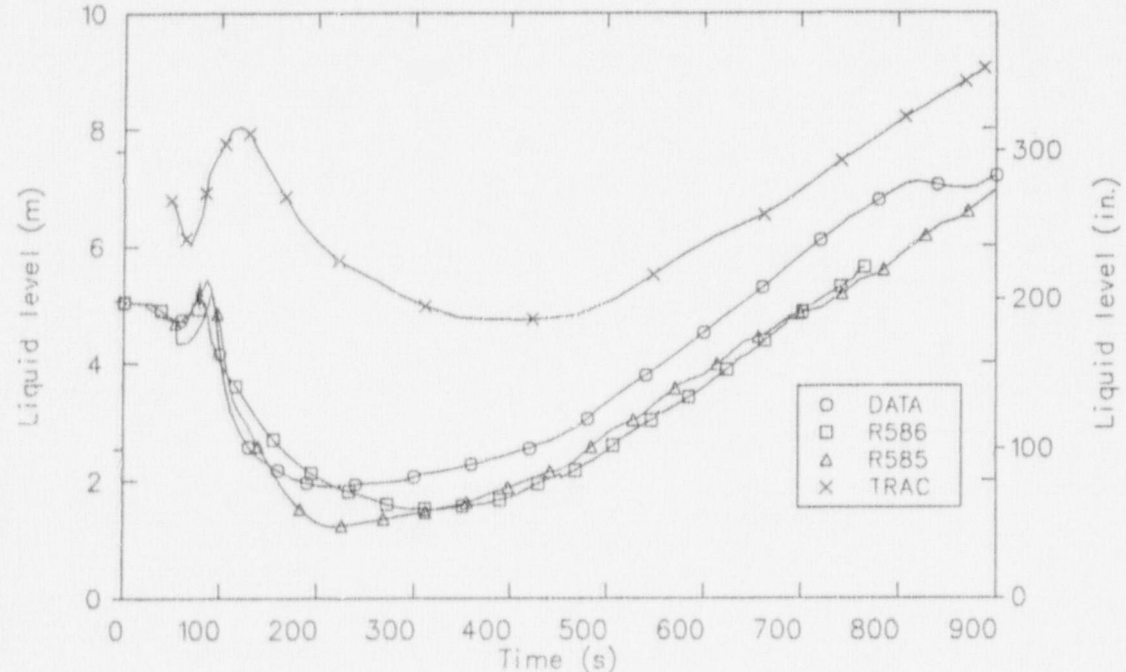

Figure 16. A comparison of calculated pressurizer liquid levels during the Davis-Besse LOFW transient.

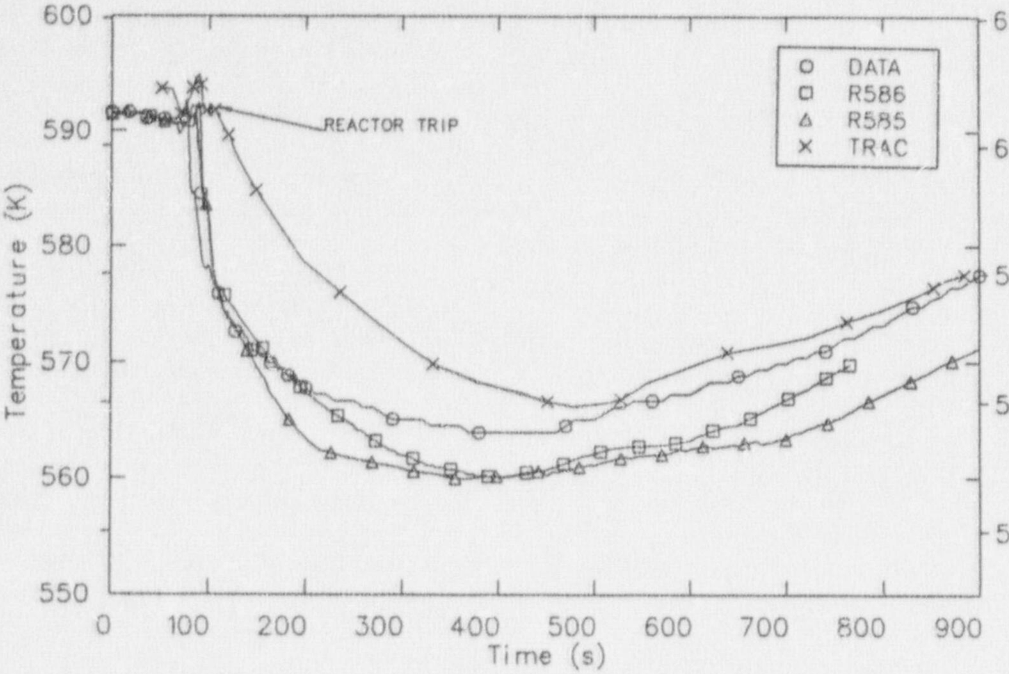

Figure 17. A comparison of calculated coolant temperatures in the B loop hot leg during the Davis-Besse LOFW transient.

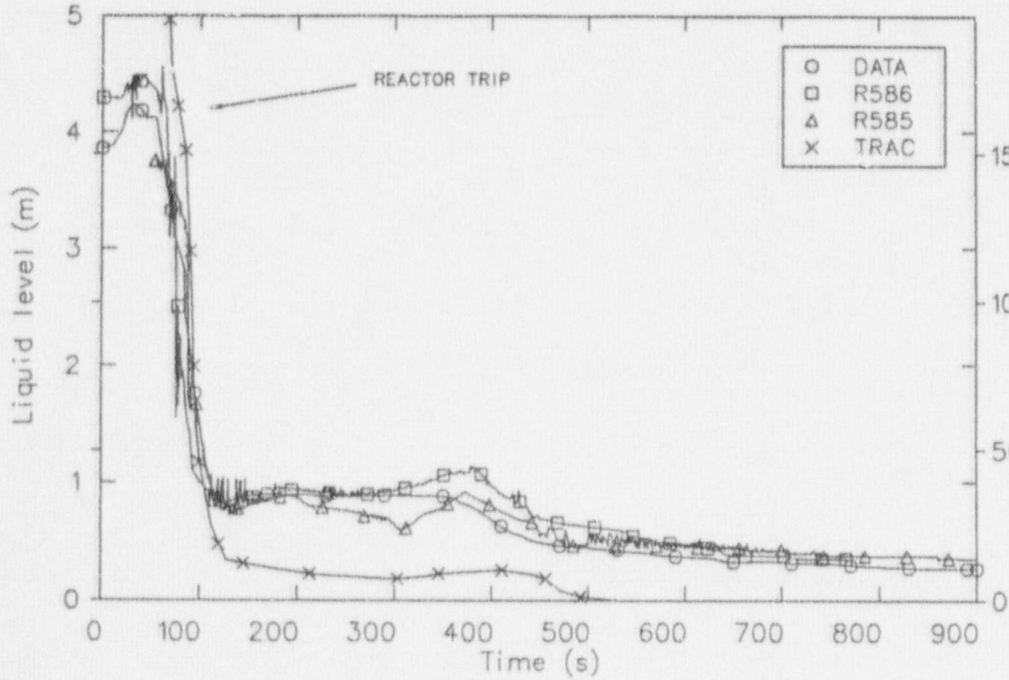

Figure 18. A comparison of calculated startup liquid levels in OTSG B during the Davis-Besse LOFW transient.

indicates that the R586 calculation had a larger feed flow, yet the RCS temperatures decreased more rapidly after reactor trip in the R585 calculation. This apparent discrepancy was attributed to the core power boundary condition. The power table in the R585 calculation went immediately to decay heat after reactor trip. The R586 curve included the effects of the fission power following reactor trip. The integrated fission power in the R586 calculation was equivalent to 1.5 s of full reactor power. The additional power in the R586 calculation increased the RCS temperature by about 3 K (6°F), equivalent to the difference between the R586 and R585 curves at 200 s. Different feedwater flows were then used to partially compensate for the different core powers so that the long-term RCS responses were similar.

Even though different codes and models were used, the results of all three calculations were similar and showed trends like those observed during the transient. The major differences between calculations were primarily due to the assumption of different core powers, feedwater flows, and pressurizer spray flows.

3.3 R586 Feed and Bleed Calculation

The R586 feed and bleed calculation was performed with the RELAP5/MOD2 model described in Section 2. The calculation is summarized in Table 4, which presents a sequence of the key events occurring in the transient. The transient was initiated from 100% rated power by tripping both MFPs at 0.0 s. Tripping both MFPs immediately caused a turbine trip and an anticipatory reactor trip.

The response of the RCS early in the transient was controlled by the secondary system. OTSG pressures, shown in Figure 19, increased following the closure of the turbine stop valves at 0.6 s. The turbine bypass valves (TBVs), AEVs, and SRVs then opened to limit the pressurization of the OTSG. The SRVs closed at 28.5 s. The ICS then operated the TBVs and AEVs to control the pressure at 7.24 MPa (1050 psia) for the remainder of the transient. The OTSG startup levels, shown in Figure 20, represented a wide range level measurement. The MFPs coasted down following the MFP trip at 0 s, providing only 0.9 s worth of full flow before check valves in the feed lines closed. Because the AFW system was assumed to fail, no source of water was available for the OTSGs. Heat transferred from the RCS boiled away the liquid inventory of the OTSGs causing the startup levels to decrease. Both OTSGs were dry by 130 s. Even when the OTSGs were dry, the indicated levels, which were based on calculated differential pressure, did not go to zero because of the weight of the steam between the differential pressure taps.

Table 4. Sequence of events for the feed and bleed calculation

Time (s)	Event
0.0	Both MFPs trip; turbine trip; reactor trip
0.60	Turbine stop valves fully closed
1.13	Pressurizer spray valve opened
2.03	TBVs and AEVs opened
2.33	OTSG SRVs opened
4.58	Pressurizer heaters on
6.00	Pressurizer spray valve closed
28.5	OTSG SRVs closed
130	OTSGs dry out
202	Pressurizer heaters off
217	Pressurizer spray valve opened
340	PORV first opened
750	Pressurizer liquid solid
1012	Reactor coolant pumps tripped
1103	Reactor vessel vent valves opened
1200	Feed and bleed initiated
1955	Pressurizer SRVs opened
3090	Last closure of the pressurizer SRVs
3752	Reactor coolant depressurization began
4707	Pressurizer spray valve closed
5162	Refill of reactor coolant system began
5500	HPI flow initiated
5800	Calculation terminated

Figure 21 shows calculated pressure in the hot leg of the A loop. The turbine trip at 0 s and the closure of the turbine stop valves caused a momentary heatup and pressurization of the reactor coolant. Pressurizer spray was initiated at 1.13 s, limiting the peak pressure to 15.5 MPa (2250 psia). The OTSGs then removed more heat from the reactor coolant than was being produced by decay heat in the reactor core, causing a decrease in pressure and hot leg temperature (see Figure 22). Once the reactor coolant temperature stabilized based on the OTSG temperature corresponding to 7.24 MPa (1050 psia), the pressure of the reactor coolant reached its minimum value and then increased slowly because of the pressurizer heaters. Core decay heat caused the reactor coolant temperature and pressure to increase rapidly after the OTSGs dried

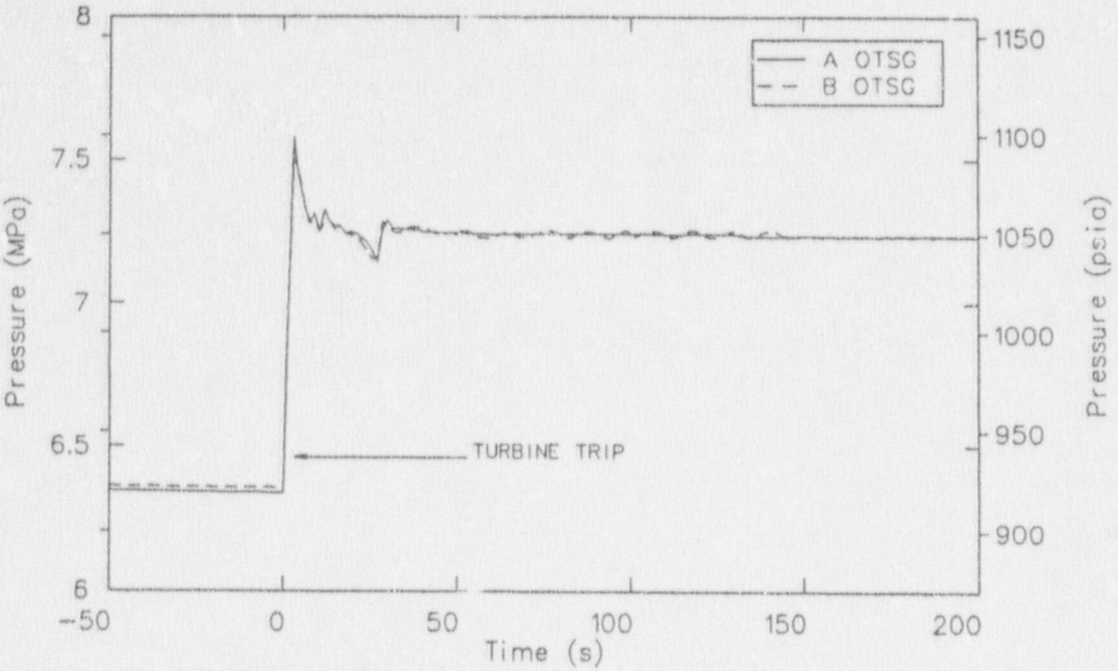

Figure 19. OTSG pressures in the R586 feed and bleed calculation.

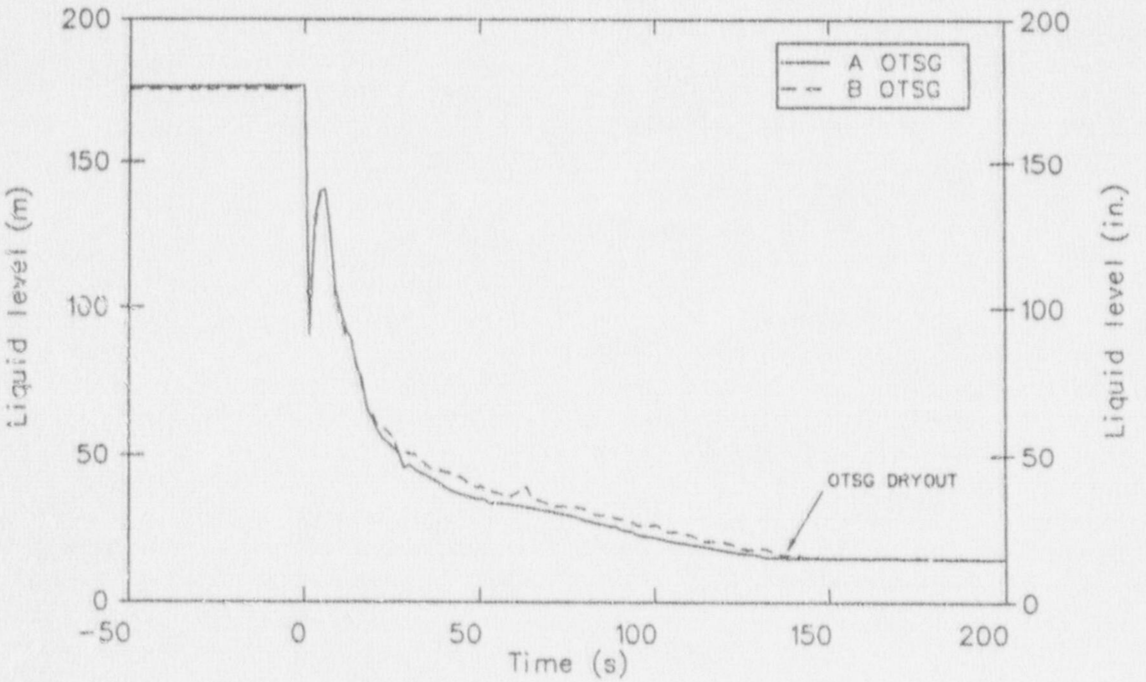

Figure 20. OTSG startup liquid levels in the R586 feed and bleed calculation.

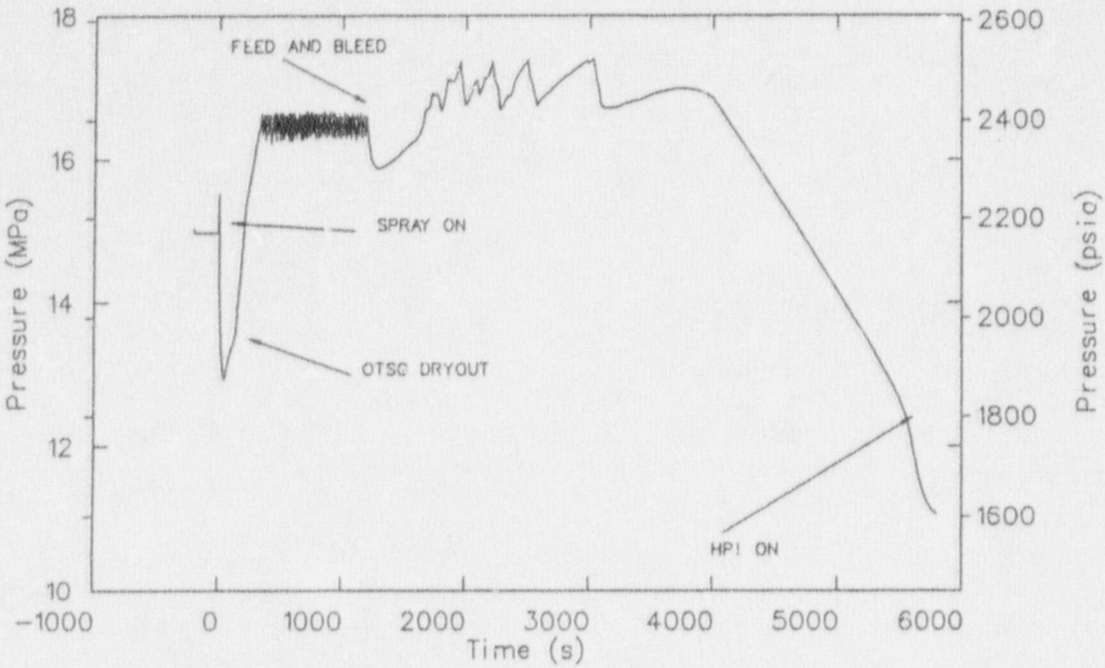

Figure 21. Reactor coolant pressure in the R586 feed and bleed calculation.

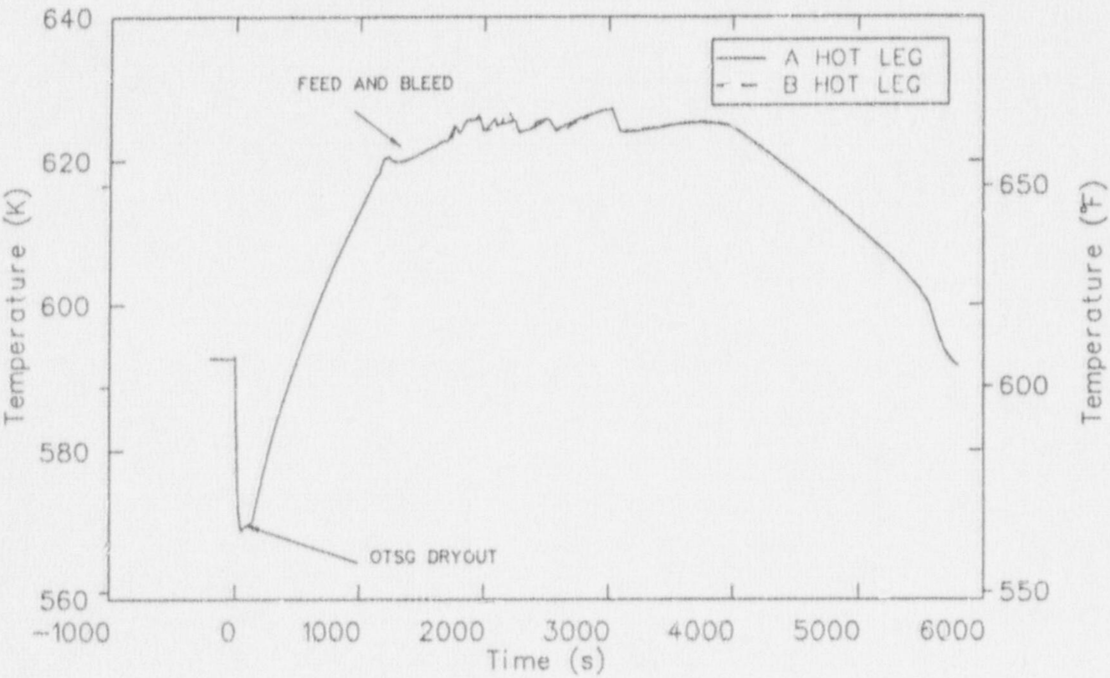

Figure 22. Hot leg coolant temperatures in the R586 feed and bleed calculation.

out at 130 s. The increase in reactor coolant pressure caused the heaters to shut off at 202 s and the spray valve to open at 217 s. The initiation of spray reduced the pressurization rate, but the PORV open setpoint pressure of 16.65 MPa (2415 psia) was reached at 340 s. The PORV then opened and closed repeatedly, maintaining the pressure between its open and close setpoints. The collapsed liquid level in the pressurizer, shown in Figure 23, increased after the OTSGs dried out because of the heatup and expansion of the reactor coolant. The liquid level reached the top of the pressurizer at 750 s, and the RCS was liquid solid. The heatup of the reactor coolant continued, and at 952 s the hot leg subcooling (see Figure 24) decreased to 11 K (20°F). The RCPs were tripped, simulating an operator action, 60 s later.

Feed and bleed cooling was initiated at 1200 s. The state of the plant at the initiation of feed and bleed cooling was as follows: the OTSGs were dry; the RCS was liquid solid; the reactor coolant pressure was near the PORV open setpoint; the hot legs were slightly [about 2 K (4°F)] subcooled; and the RCPs were tripped. Feed and bleed cooling was initiated by locking open the PORV and HPVVs, starting both makeup pumps, starting the HPI pumps and aligning them to take suction from the discharge of the LPI pumps, and tripping the pressurizer heaters.

The reactor coolant pressure (recall Figure 21) rapidly decreased about 0.7 MPa (100 psi) after the PORV was locked open. However, the decrease in pressure eliminated the subcooling in the hot legs and allowed saturated boiling in the core. The PORV, which was passing subcooled liquid, was not able to relieve the volumetric expansion due to the boiling in the core; and the reactor coolant repressurized. The pressure increased to 17.34 MPa (2515 psia), the setpoint of the pressurizer SRVs, at 1955 s, and the SRVs opened. The open SRVs were able to depressurize the system until the pressure dropped low enough to allow the valves to reseat. The SRVs opened and closed four times between 1955 and 3090 s, as shown in Figure 25.

The mass balance of the RCS is summarized in Figure 26, which presents PORV, HPVV, makeup, and HPI mass flow rates. The figure reveals that the combined flow through both hot leg HPVVs was insignificant compared to the flow through the PORV. The PORV flow was also several times larger than the makeup flow at the initiation of feed and bleed. The mass loss caused voids to form in the two-phase RCS, resulting in mixture levels in the vertical components.

Figure 27 presents collapsed liquid levels in the hot legs. The collapsed liquid level in the A loop hot leg dropped to the elevation of the pressurizer surge line at 2560 s. Thereafter, more steam passed through the

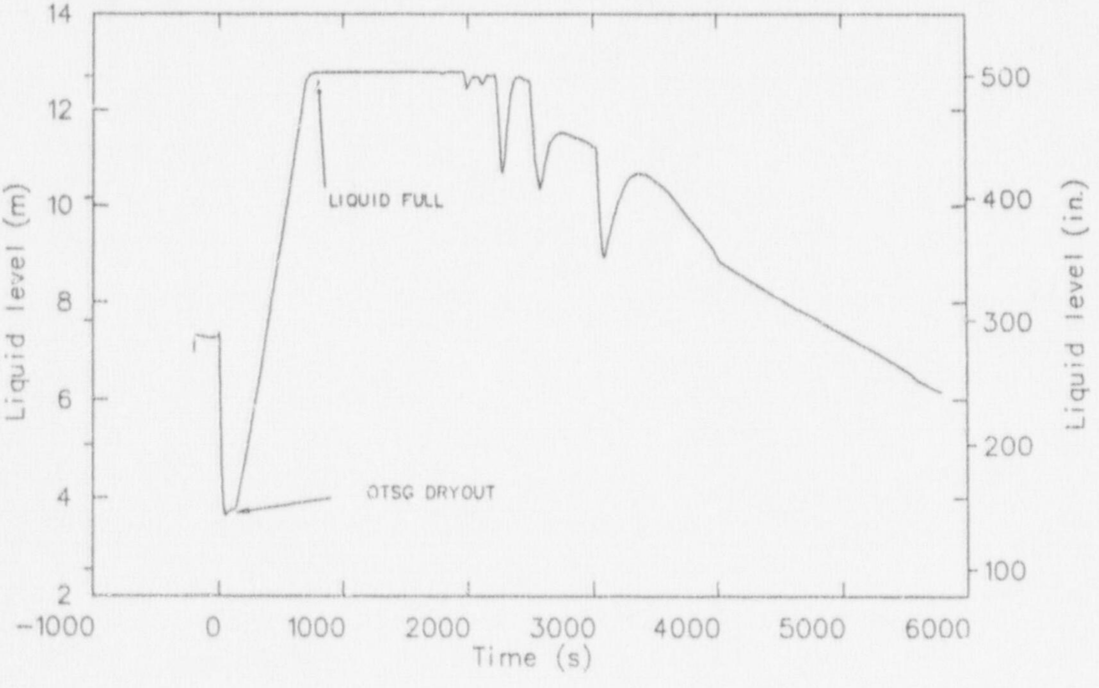

Figure 23. Pressurizer liquid level in the R586 feed and bleed calculation.

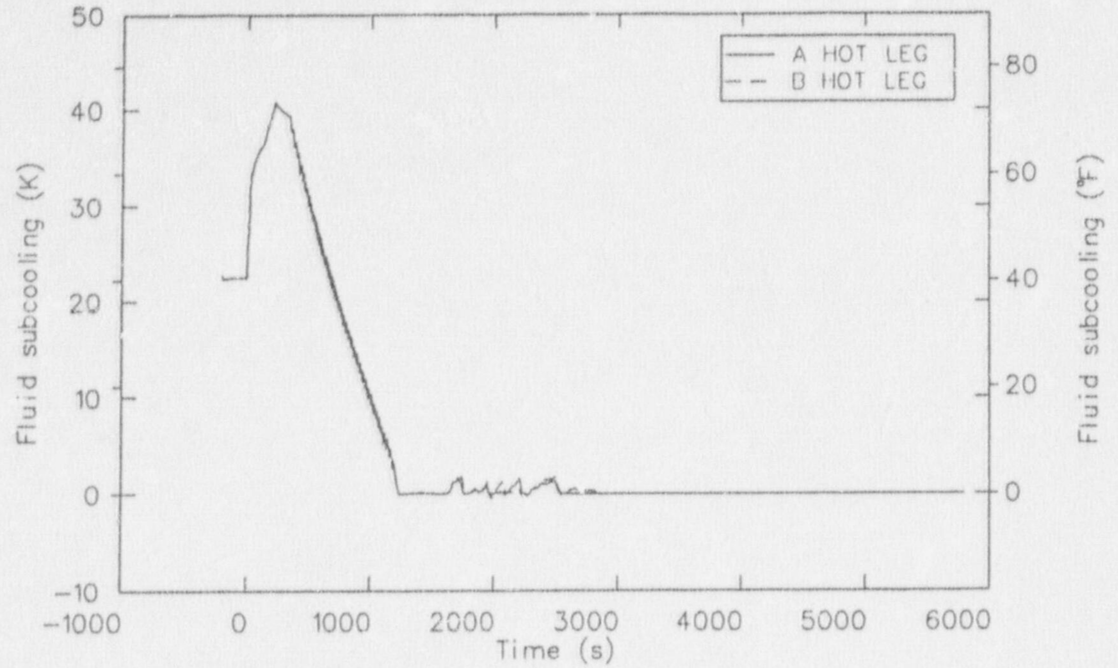

Figure 24. Hot leg subcooling in the R586 feed and bleed calculation.

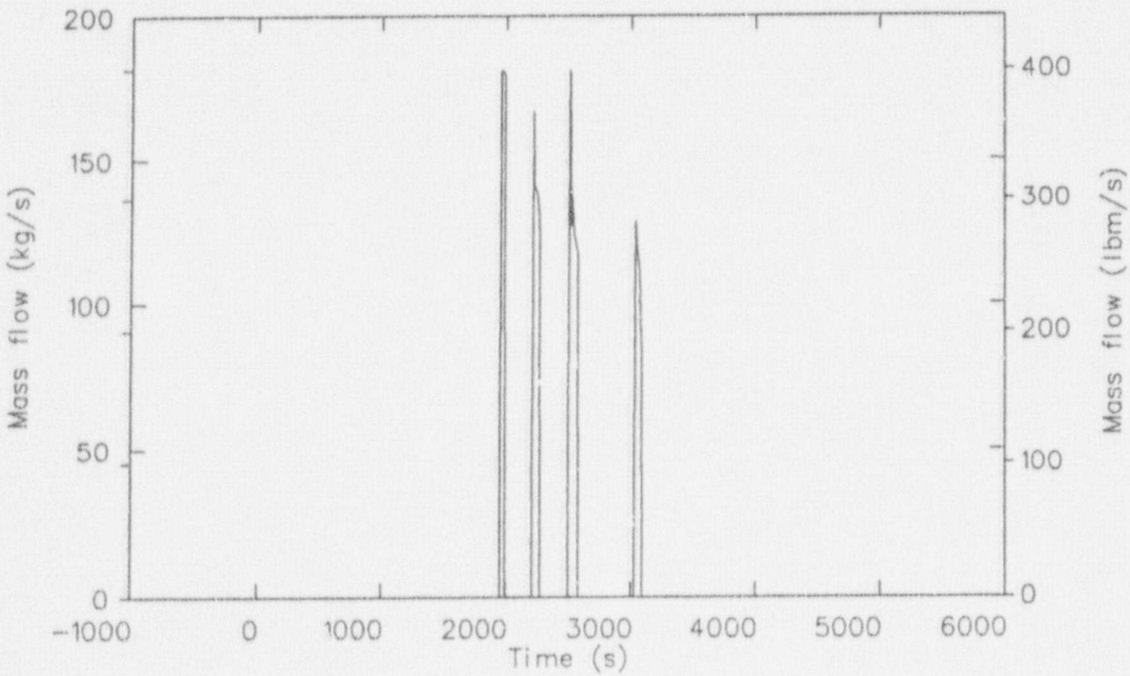

Figure 25. Mass flow through the pressurizer SRVs in the R586 feed and bleed calculation.

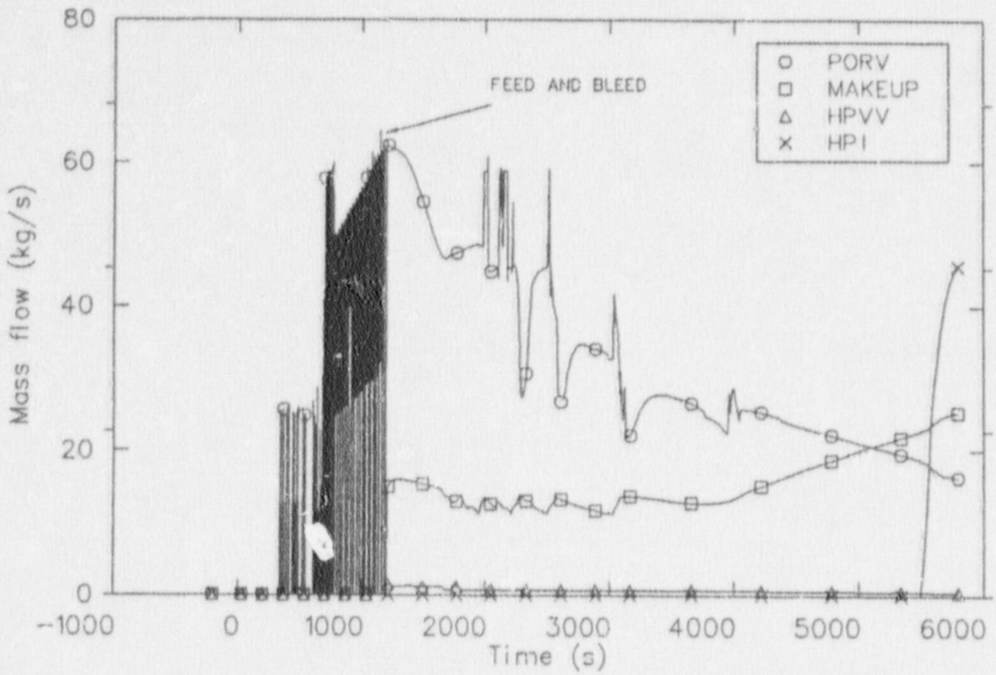

Figure 26. A summary of mass flows into and out of the RCS in the R586 feed and bleed calculation.

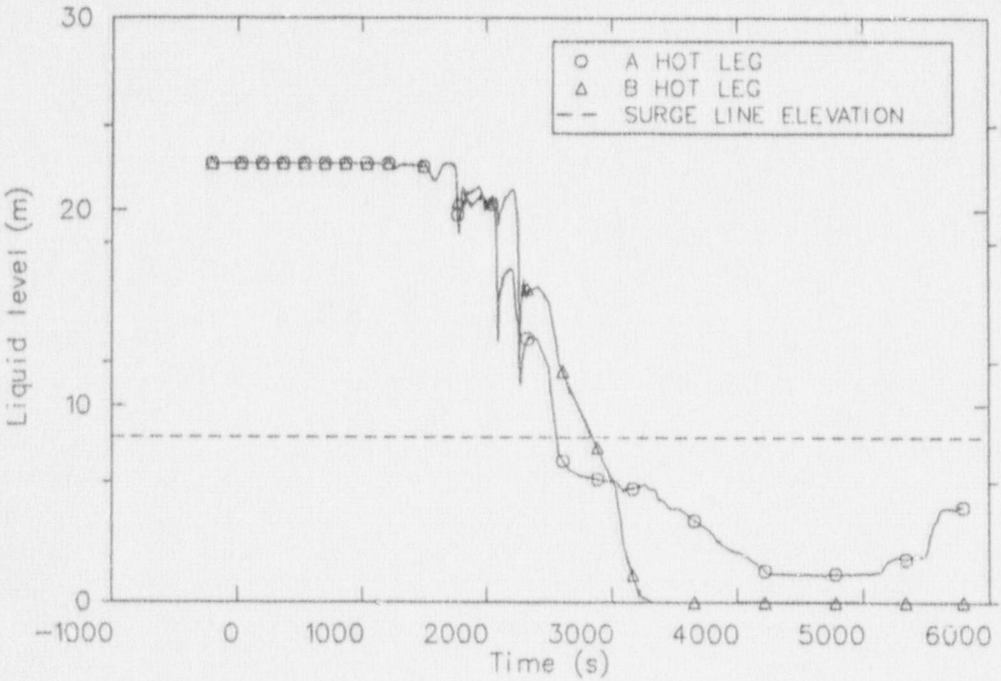

Figure 27. Hot leg collapsed liquid levels in the R586 feed and bleed calculation.

surge line and out the PORV. The increase in steam at the PORV is illustrated in Figure 28, which presents void fraction in the hot leg volume connected to the surge line and in the top volume of the pressurizer which was connected to the PORV. The void fraction at the connection to the surge line and the PORV increased significantly after the liquid level dropped to the elevation of the surge line. As expected, there was a strong correlation between the void fraction in the hot leg and the void fraction at the PORV.

The increase in void fraction at the PORV resulted in an increased volumetric flow out the PORV. By 3752 s, the volumetric flow out the PORV was large enough to depressurize the RCS, as shown in Figure 21. The reactor coolant pressure decreased for the remainder of the calculation. The increase in void fraction and the decrease in pressure caused the flow out the PORV to decrease (see Figure 26). The decreasing pressure also resulted in an increasing makeup flow. The makeup flow exceeded the combined flow out the PORV and the HPVVs after 5162 s, beginning a gradual refill of the RCS. The pressure dropped below the HPI shutoff head of 12.65 MPa (1835 psia) at 5500 s. The addition of HPI significantly increased the flow into and the refill rate of the RCS.

The liquid inventory in the reactor vessel is shown in Figure 29. The plot represents the collapsed level between the bottom of the lower reactor head, through the core and the upper plenum, to the top of the upper reactor head. The upper head began draining about 200 s after the initiation of feed and bleed, resulting in a decreasing liquid level. The liquid level decreased rapidly until it dropped below the hot leg nozzles. The level then decreased slowly until makeup began refilling the RCS. The level increased rapidly after HPI was initiated. The minimum collapsed liquid level was about 0.3 m (1 ft) above the top of the core. In fact, the calculated mixture (froth) level never dropped below the hot legs nozzles. The core was covered with liquid or a two-phase mixture throughout the transient. Consequently, no core heatup was calculated and the fuel rod cladding temperatures stayed within a few degrees of the fluid temperature.

The calculation was terminated at 5800 s. Hand calculations indicated that a quasi-steady state would be achieved at a pressure near 8.6 MPa (1250 psia) with subcooled liquid exiting the PORV. At this pressure, a mass and energy balance could be achieved. The flow out the PORV and HPVVs would balance makeup and HPI. The core power would heat the injected water, which would then flow out the open valves. After this quasi-steady state was obtained, the pressure would drop slowly as the decay heat decreased. A source of feedwater would be required to ultimately bring the plant to cold shutdown.

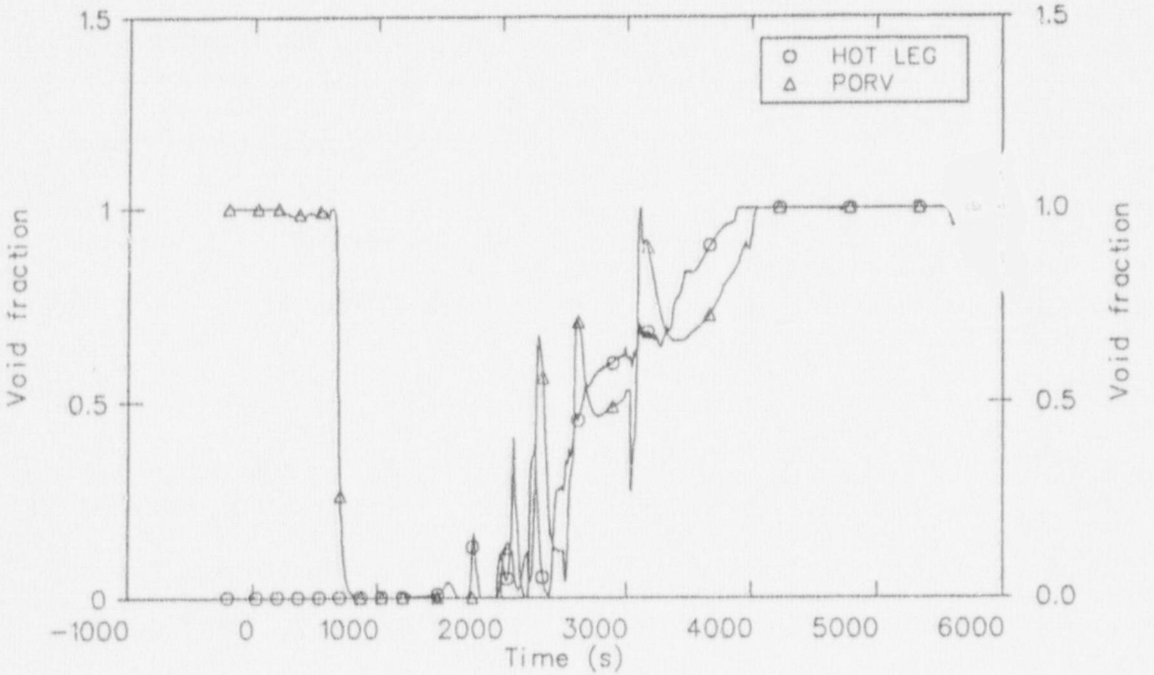

Figure 28. Hot leg and PORV void fractions in the R586 feed and bleed calculation.

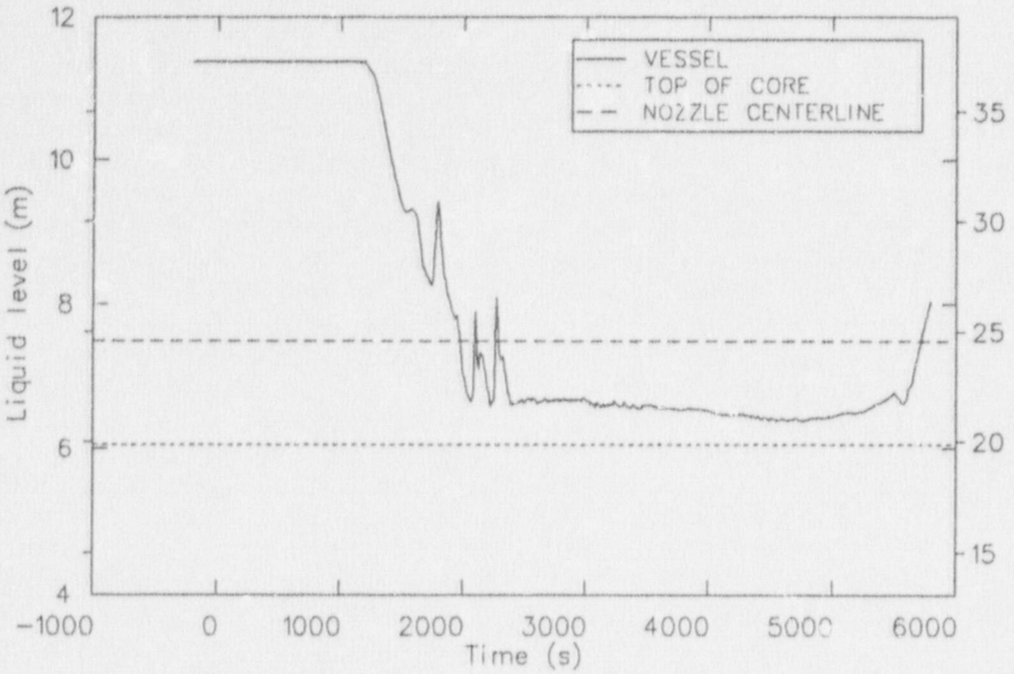

Figure 29. Collapsed liquid level in the reactor vessel during the R586 feed and bleed calculation.

3.4 Comparison of Feed and Bleed Calculations

Feed and bleed calculations have been performed for Davis-Besse at two different power levels. Section 3.4.1 contains a comparison of the R585 and R586 feed and bleed calculations for a LOFW transient initiated from 100% power. A comparison of the R585 and TRAC feed and bleed calculations for a LOFW transient initiated from 90% power are compared in Section 3.4.2.

3.4.1 100% Power. The R586 feed and bleed calculation described in Section 3.3 was a repeat of one of the R585 calculations performed shortly after the June 9, 1985, LOFW event at Davis-Besse. The determination of the uncertainty in the R585 calculations was the major purpose of this study. The following comparison of the R586 and R585 calculations was used in the determination of the uncertainty in the calculations. The major difference between calculations was that the R586 calculation was performed with a quality-assured model of Davis-Besse, while the R585 calculation was performed with a model based on Oconee Unit 1 that was quickly modified by INEL to resemble Davis-Besse.

Figure 30 presents a comparison of hot leg pressures from the R585 calculation, the R586 base calculation (described in Section 3.3), and a sensitivity calculation which will be discussed later. Several differences were identified between the R585 and R586 base calculations. First, an earlier dryout of the OTSGs was obtained in the R586 calculation, 130 s versus 220 s in the R585 calculation. The different dryout times were primarily caused by differences in the amount of feedwater delivered to the OTSGs following MFP trip. In the R586 calculation, an amount of feedwater corresponding to 0.9 s of steady-state flow was delivered to the OTSGs as the MFPs coasted down. In the R585 calculation, 2.3 s of steady-state flow were delivered. Since the R586 model represented the Davis-Besse feedwater system and not the Oconee system, the R586 calculation better represented the OTSG dryout time for Davis-Besse.

Second, the reactor coolant pressure increased more rapidly in the R586 calculation than in the R585 calculation because of the earlier OTSG dryout time and the smaller pressurizer spray flow rate described in Section 3.2. Consequently, the PORV open setpoint pressure was reached at 340 s in the R586 calculation versus 620 s in the R585 calculation.

Third, the pressure decreased less before the onset of boiling in the core after the initiation of feed and bleed in the R586 calculation. This was an indication that the hot leg temperature was higher in the R586 calculation. The higher hot leg temperature was caused by the earlier dryout of the OTSG and a slightly higher core decay power in the R586 calculation. The core power in the R586 calculation was augmented to account for actinide decay.

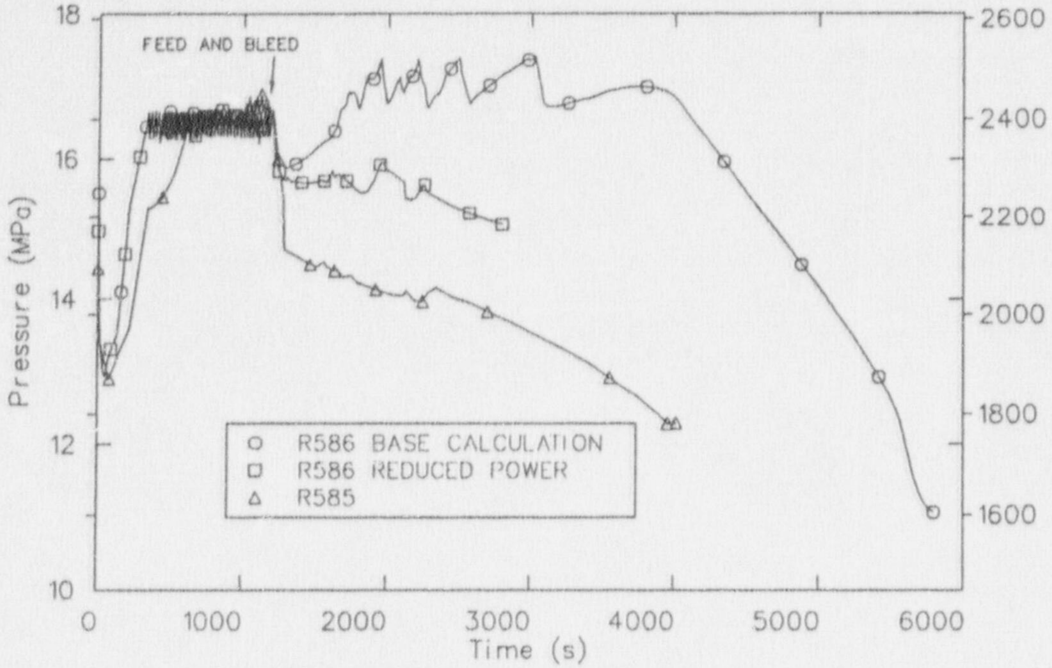

Figure 30. A comparison of calculated reactor coolant pressures during feed and bleed.

Finally, a major difference in the calculated pressures was observed following the initiation of feed and bleed cooling. The pressure increased following the onset of boiling in the core in the R586 calculation, while the pressure decreased in the R585 calculation. The deviation between calculations was caused by the core power boundary condition. An error was discovered in the R585 model which reduced the core power 25% after the RCPs tripped. The model, as developed by INEL, represented the ICS runback of reactor power following the trip of both RCPs in one loop. This runback is applicable only prior to reactor trip but was inadvertently applied to the core decay power after reactor trip.

A sensitivity calculation was performed in which the R586 power was reduced 25% at the initiation of feed and bleed, as shown in Figure 31. The effect of the reduced power on the reactor coolant pressure is also shown in Figure 30. The R586 sensitivity calculation depressurized at about the same rate as the R585 calculation. The pressure remained higher because of the difference in hot leg temperature at the initiation of feed and bleed, as discussed previously. The sensitivity calculation showed that the major difference between calculations was caused by the power boundary condition.

Pressurizer collapsed liquid levels from the R585, R586 base, and R586 sensitivity calculations are shown in Figure 32. The pressurizer filled with liquid earlier in the R586 calculations because of the earlier OTSG dryout discussed previously. The pressurizer level dropped more rapidly in the R586 base calculation after feed and bleed was initiated because the higher core power generated more steam which flowed to the pressurizer. The levels responded similarly in the R585 and sensitivity calculations.

The reactor coolant mass balance is summarized in Figures 33 and 34. Figure 33, which shows the combined PORV and HPVV flows, reveals that the total flow out of the RCS was similar in all three calculations. The total flow into the system, shown in Figure 34, was determined by the reactor coolant pressure. The R585 calculation had the lowest pressure and thus the highest makeup flow after the initiation of feed and bleed.

A comparison of Figures 33 and 34 shows that the RCS contained more mass in the R585 calculation. The higher system mass in the R585 calculation resulted in a higher collapsed liquid level in the reactor vessel, as shown in Figure 35. Part of the difference in vessel levels was caused by differences in the noding of the upper plenum. The R586 model explicitly represented the small holes in the plenum cylinder at the hot leg nozzle elevation. Modeling these holes allowed a more accurate representation of the flow paths in the upper plenum, the draining from the hot legs, and the mixture level in the vessel.

The comparison of the R585 and R586 feed and bleed calculations indicates that the transient response, and in particular the reactor coolant pressure, varied

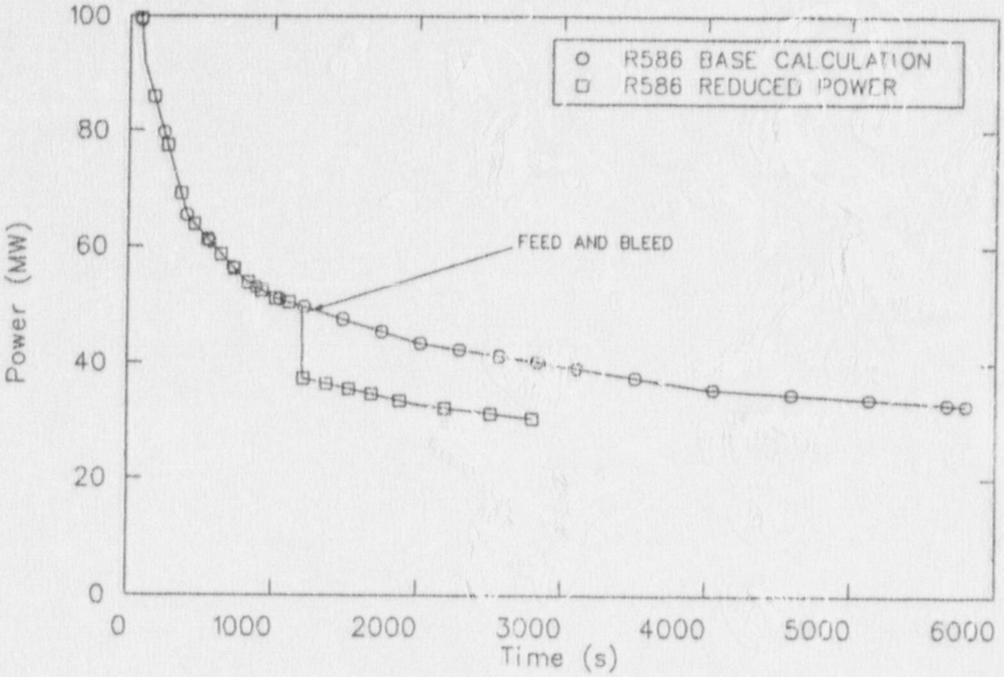

Figure 31. A comparison of core power from the R586 base and sensitivity feed and bleed calculations.

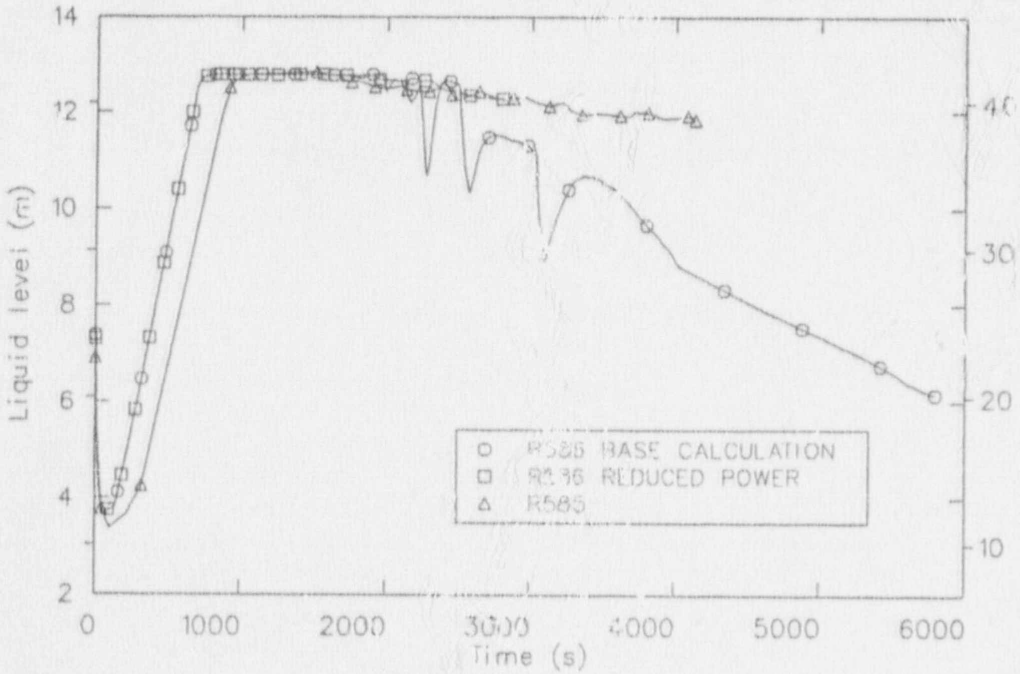

Figure 32. A comparison of calculated collapsed pressurizer liquid levels during feed and bleed.

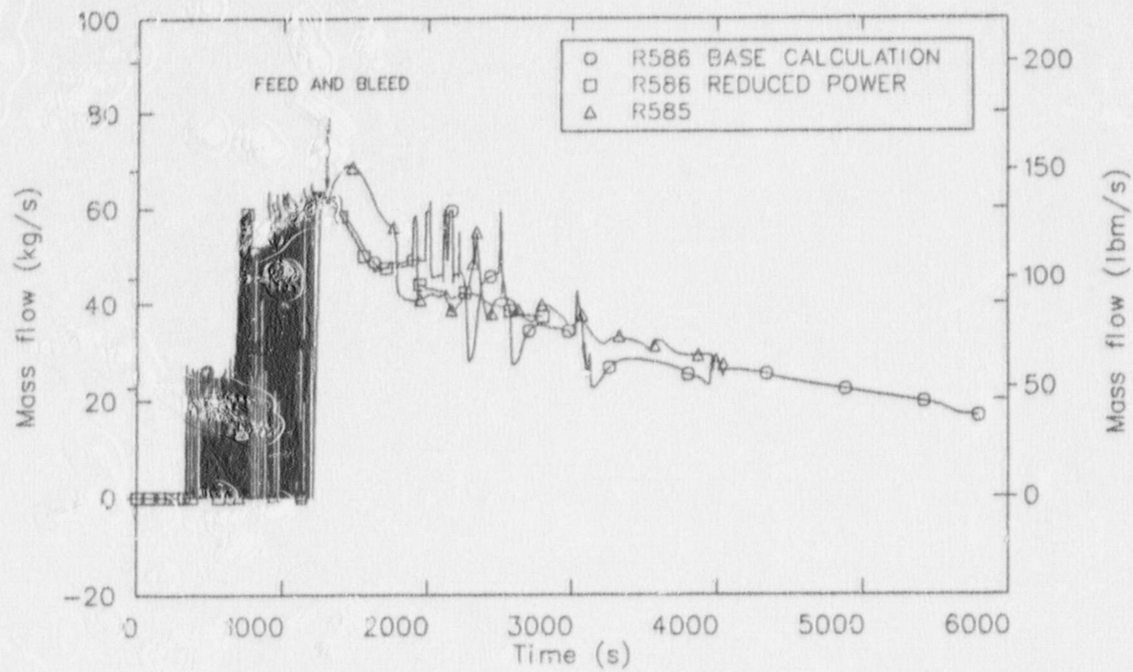

Figure 33. A comparison of calculated mass flows out of the RCS during feed and bleed.

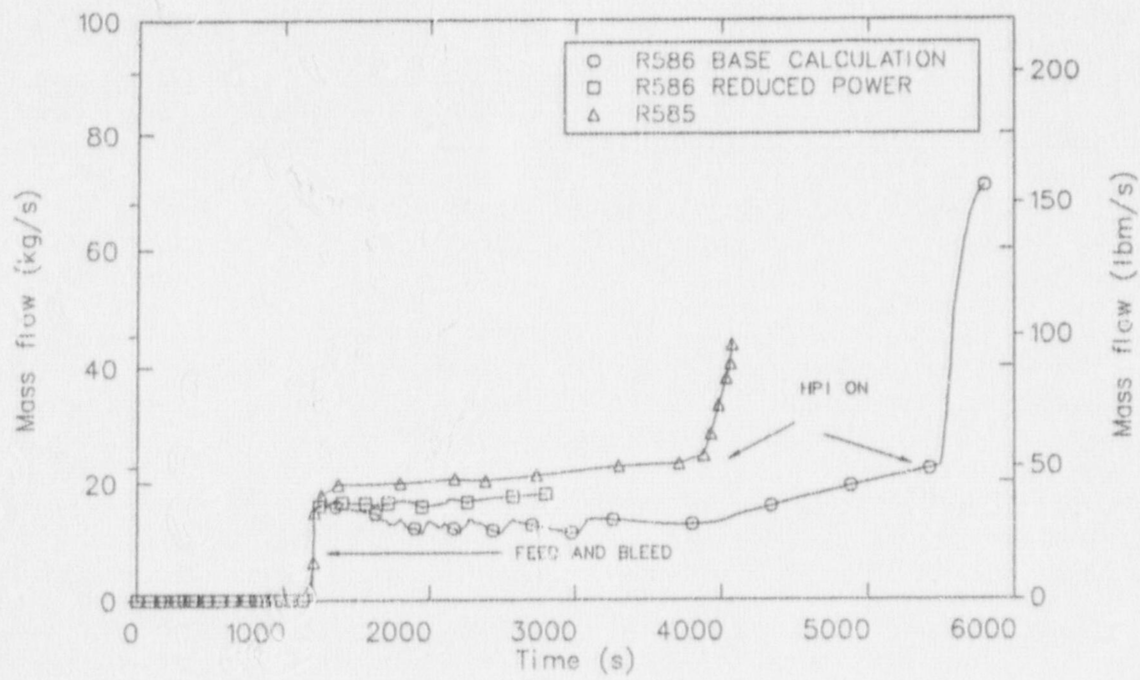

Figure 34. A comparison of calculated mass flows into the RCS during feed and bleed.

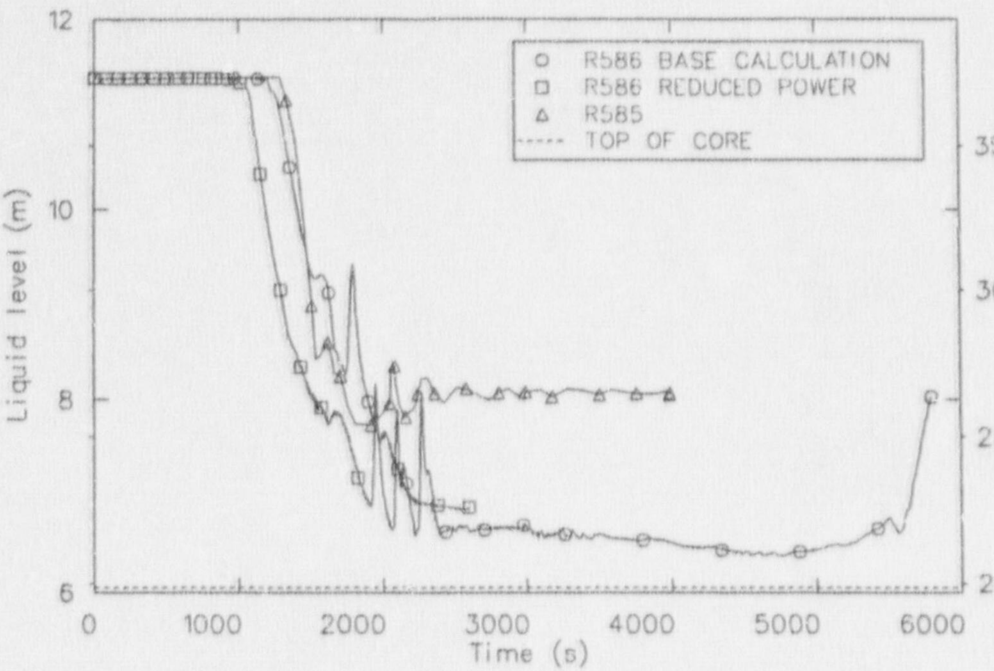

Figure 35. A comparison of calculated collapsed liquid levels in the reactor vessel during feed and bleed.

substantially. Timing of events also varied significantly. Differences between the calculations were primarily attributed to the core power boundary condition. However, in a macroscopic sense, the two calculations provided similar behavior. Both the R586 and R585 calculations indicated that feed and bleed would successfully depressurize the RCS while adequately cooling the core.

3.4.2 90% Power. The R585 and TRAC feed and bleed calculations from 90% power were parametric variations of the LOFW transient of June 9, 1985, at Davis-Besse. These calculations were similar to the calculations described in Section 3.4.1 except that the initial power was lower and one MFP was not tripped immediately, thus resulting in the delivery of more feedwater to the OTSGs. NRR and LANL performed two comparable feed and bleed calculations. The feed and bleed calculations were initiated at hot leg temperatures near 617 K (650°F) or 589 K (600°F).

In the first comparison, feed and bleed was initiated near 2400 s in the R585 calculation and near 2000 s in the TRAC calculation. The corresponding hot leg temperatures were near 617 K (650°F) and 609 K (636°F), respectively. Hot leg pressures from the R585 and TRAC calculations are compared in Figure 36. The pressures were similar prior to feed and bleed except for the effects of the larger pressurizer spray in the R585 calculation, as described in Section 3.2. After the initiation of feed and bleed, the pressures dropped rapidly until the subcooling in the hot legs vanished and saturated boiling began in the core. In the R585 calculation, the pressure then continued to decrease, but at a slower rate. HPI was initiated at 4200 s, and the core was successfully cooled by feed and bleed. In the TRAC calculation, however, the reactor coolant repressurized slowly. The coolant began to depressurize near the end of the TRAC calculation because of slight voiding at the PORV. Although the calculation was not taken to HPI initiation, a LANL extrapolation indicated that feed and bleed would successfully cool the the core.

The large difference in the pressure response between the TRAC and R585 calculations was attributed to the boundary conditions. The R585 calculation depressurized too quickly because of the erroneous 25% reduction in core decay power when the RCPs were tripped. This error was discussed in the previous section. The TRAC calculation depressurized too slowly because of a smaller-than-realistic PORV flow.

Figure 37 shows the PORV flow from the TRAC calculation and the combined PORV and HPVV flows from the R585 calculation. The TRAC and R585 curves are comparable because the HPVV flow was generally insignificant compared to the PORV flow. The figure indicates that the PORV flow from the R585 calculation was significantly higher than the TRAC flow, even though the reactor coolant pressure was generally higher in the TRAC calculation. A review of data indicates that the TRAC PORV flow was

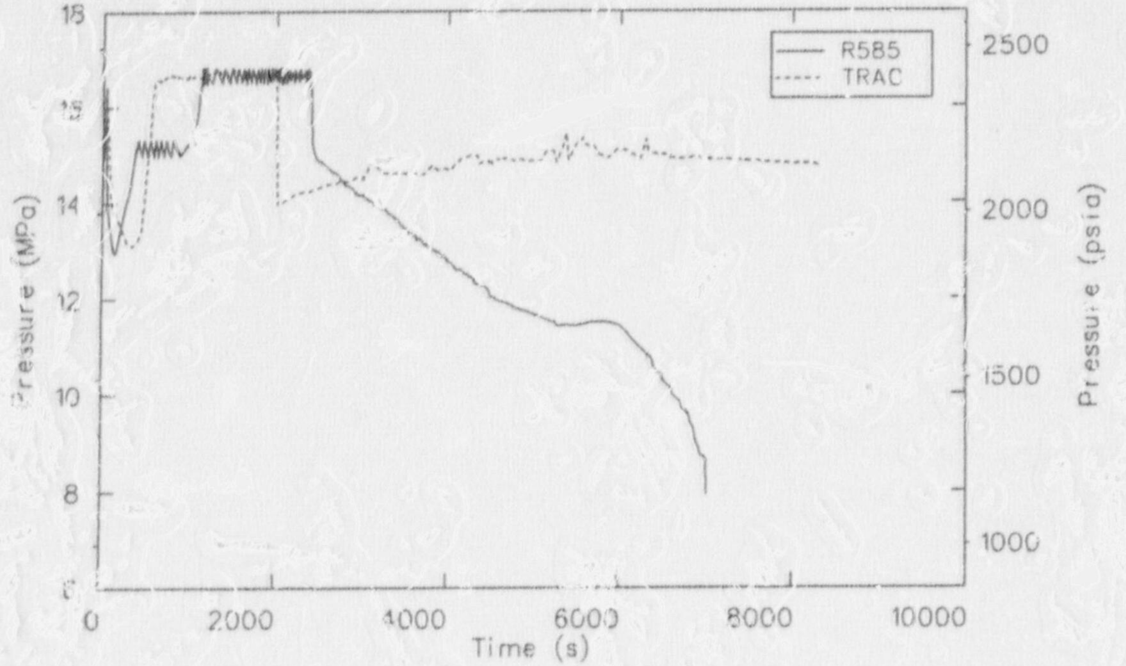

Figure 36. A comparison of calculated reactor coolant pressures when feed and bleed was initiated near 2200 s

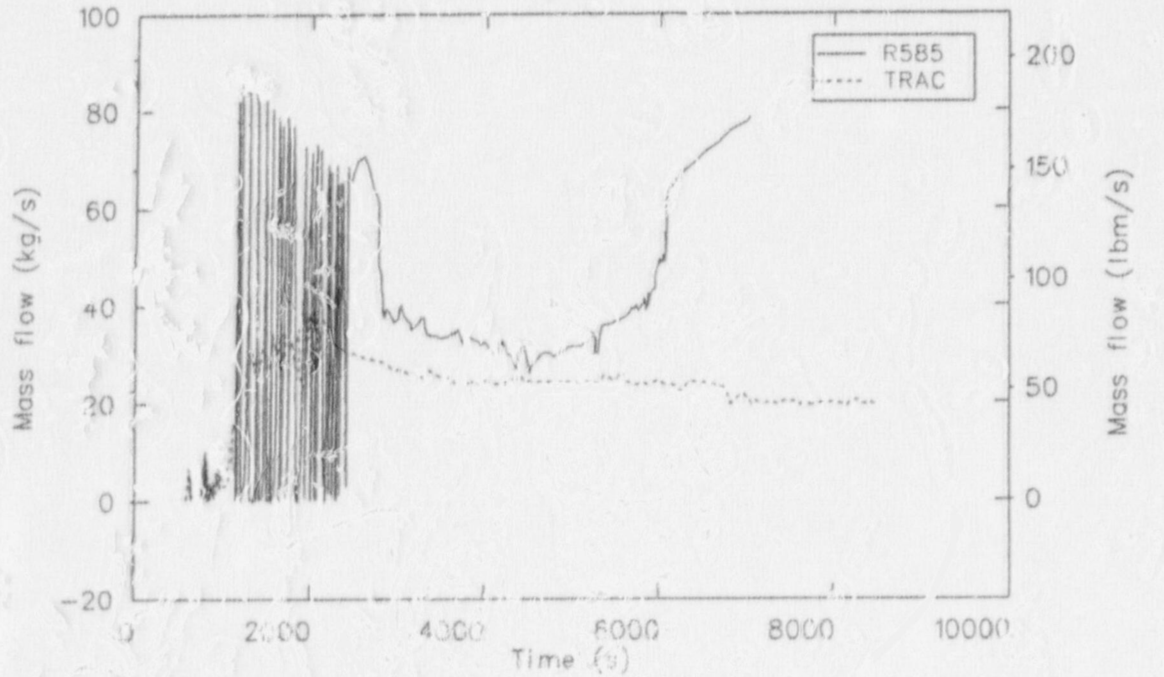

Figure 37. A comparison of calculated mass flow out of the RCS when feed and bleed was initiated near 2200 s.

significantly too small. For example, at 5000 s in the TRAC calculation, the PORV was passing liquid at a rate of about 24 kg/s (53 lbm/s) at a pressure of about 14.8 MPa (2150 psia). The Electric Power Research Institute (EPRI) conducted critical flow tests through a Crosby PORV similar to the one in Davis-Besse. The EPRI data[9] indicated that for liquid flow at 16.5 MPa (2390 psia), the EPRI PORV would pass about 47 kg/s (104 lbm/s). It is estimated that at the TRAC pressure of 14.8 MPa (2150 psia), the PORV should have passed about 43 kg/s (95 lbm/s). This estimate was obtained by adjusting the EPRI flow rate to account for the 9% larger bore diameter in the Davis-Besse PORV compared to the valve tested by EPRI and to account for the different fluid conditions in the TRAC calculation and the EPRI test. Thus, it appears that the TRAC PORV flow rate is about a factor of two low for liquid. Consequently, the RCS voided too slowly in the TRAC calculation. Since the depressurization of the reactor coolant was coupled to voiding at the PORV, the TRAC calculation depressurized too slowly. The correct pressure response lies between the R585 and TRAC curves shown in Figure 36.

The R585 and TRAC calculations generally were in good agreement for parameters other than the reactor coolant pressure and PORV flow. The R585 and TRAC fluid subcoolings for the A loop hot leg are compared in Figure 38. The two calculations were in close agreement until subcooling was recovered near the end of the R585 calculation. The collapsed liquid level remained at or near the top of the pressurizer after feed and bleed was initiated in both calculations, as shown in Figure 39. The collapsed liquid level in the vessel remained above the top of the core in both calculations, as shown in Figure 40. The liquid level initially decreased more rapidly in the R585 calculation because of the larger PORV flow. However, the pressure dropped below the HPI shutoff head after 4200 s in the R585 calculation, allowing HPI flow, and causing an increase in vessel liquid level.

NRR and LANL also performed calculations in which feed and bleed was initiated earlier and at lower hot leg temperatures than the calculations described above. In the R585 calculation, feed and bleed was initiated near 1600 s at a hot leg temperature of 594 K (610°F). In the TRAC calculation, feed and bleed was initiated near 1100 s at a hot leg temperature of about 586 K (595°F). A comparison of calculated reactor coolant pressures is presented in Figure 41. The pressure dropped rapidly in the R585 calculation after the initiation of feed and bleed and then leveled out just below the HPI shutoff head. The pressure decreased less rapidly in the TRAC calculation because of the smaller PORV flow rate discussed previously.

In the previous discussion of Figure 36, in which feed and bleed was delayed compared to the current

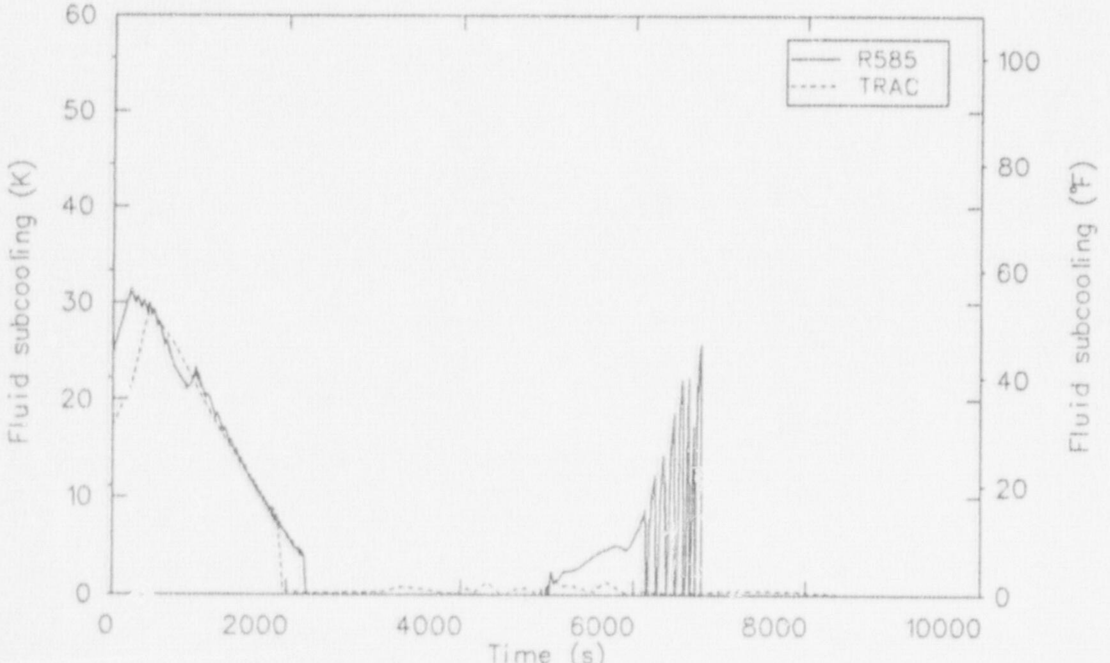

Figure 38. A comparison of calculated subcooling in the A loop hot leg when feed and bleed was initiated near 2200 s.

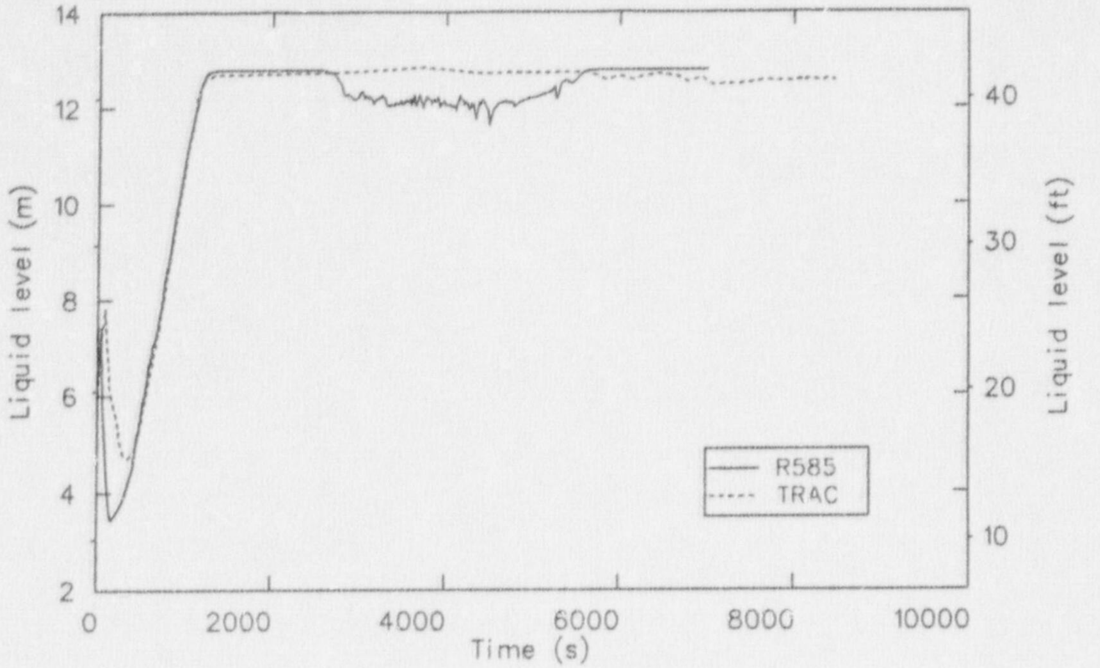

Figure 39. A comparison of calculated collapsed liquid levels in the pressurizer when feed and bleed was initiated near 2200 s.

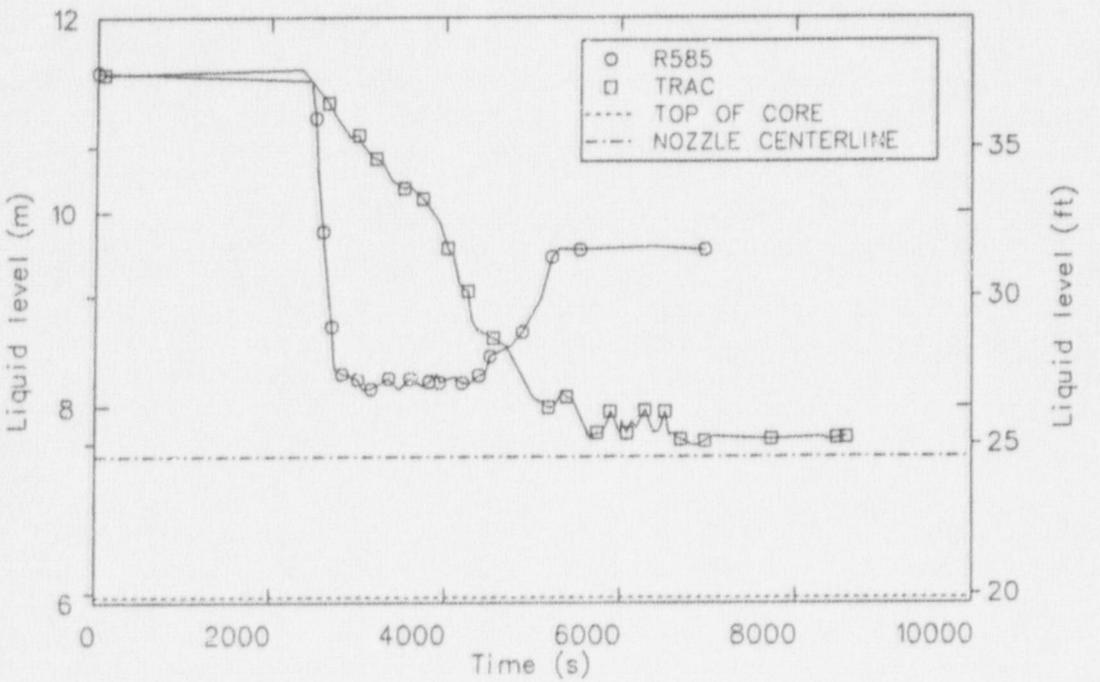

Figure 40. A comparison of calculated collapsed liquid levels in the reactor vessel when feed and bleed was initiated near 2200 s.

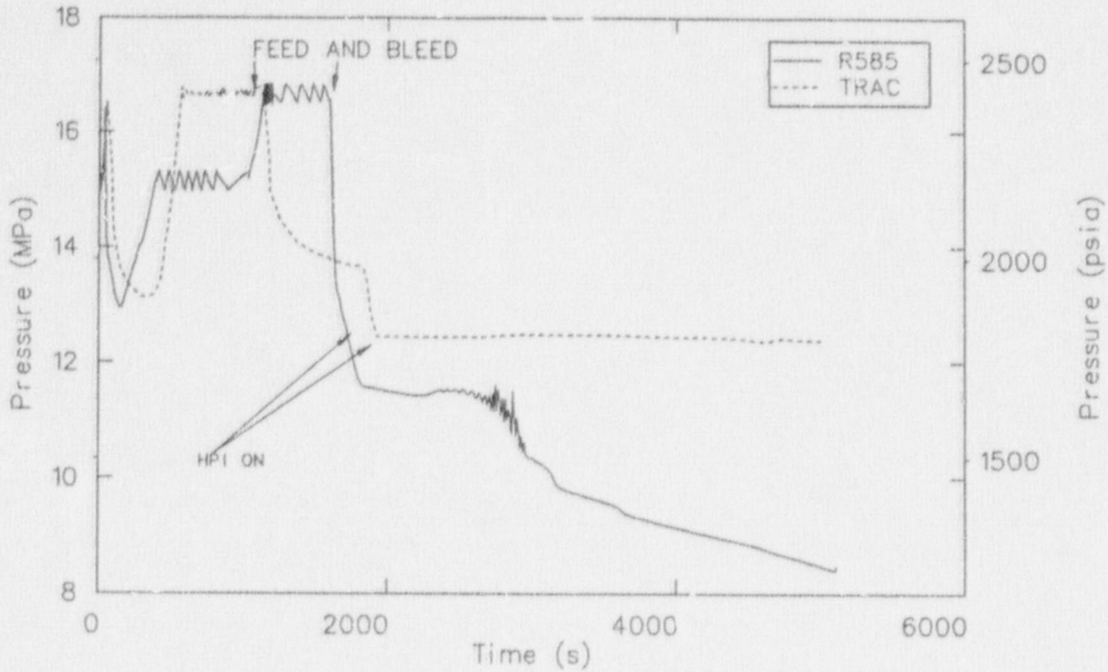

Figure 41. A comparison of calculated reactor coolant pressures when feed and bleed was initiated near 1500 s.

calculations, the rapid initial depressurization was limited by flashing and boiling in the core. In the current calculations, the depressurization was halted by the HPI, which nearly balanced the flow out the PORV. As discussed previously, the core power was too low after the RCP trip in the R585 calculation and the PORV flow was too low in the TRAC calculation. Consequently, the R585 calculation depressurized too rapidly, and the TRAC calculation depressurized too slowly. The actual pressure response should lie between the two calculated results.

Figure 42 shows the HPI flow rates in the TRAC and R585 calculations. HPI flow was initiated at 1700 and 1900 s in the R585 and TRAC calculations, respectively. The HPI flow was much higher in the R585 calculation because of the lower reactor coolant pressure shown in Figure 41. The HPI helped maintain subcooling in the RCS, as illustrated by Figure 43, which shows subcooling in the A loop hot leg. As shown in the figure, the A loop hot leg remained subcooled throughout both calculations.

In the TRAC calculation, the RCS remained liquid solid, as illustrated by Figures 44 and 45, which show collapsed liquid levels in the pressurizer and the reactor vessel. In the R585 calculation, the core and the A loop stayed subcooled, while some voiding occurred in the pressurizer, the B loop, and the upper head after the initiation of feed and bleed. The voiding in the upper head caused the collapsed level in the vessel to decrease. The difference in the vessel level response between the R585 and TRAC calculations was attributed to the upper head model. In the R585 model, the upper head was modeled as a stagnant volume, with one flow connection to the upper plenum. Consequently, the temperature in the upper head remained nearly constant until the pressure dropped low enough for it to flash and void. In the TRAC model, the upper head was modeled as a flow-through volume with two flow connections, one to the upper plenum and one to the control rod guide tube brazements which were also connected to the upper plenum. The flow circulated between the upper head and the upper plenum, coupling the temperatures of the two regions. The upper head thus remained subcooled in the TRAC calculation.

Note that the R586 model also used a flow-through upper head similar to the TRAC model. The actual mixing process that would occur in the upper head for this transient is not known. The lower 40% of the upper head is below the top of the guide tube brazements and should mix well with the upper plenum fluid. However, the mixing that would occur in the upper 60% of the upper head is governed by a relatively complicated multidimensional natural circulation process, which was not mechanistically represented in any of the models. Although the actual mixing process is not well understood, the stagnant and flow-through models bound the possible responses. It is felt that the flow-through model provides the best representation of the upper head for most transients. The possible formation of a steam bubble in the upper head leads

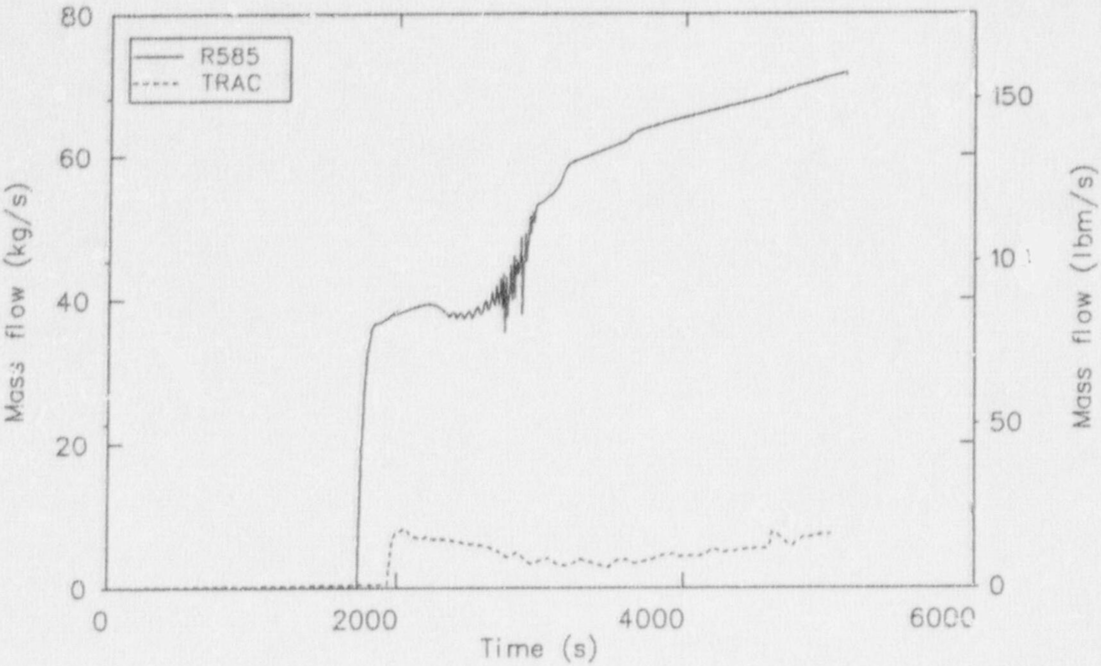

Figure 42. A comparison of calculated HPI flow rates when feed and bleed was initiated near 1500 s.

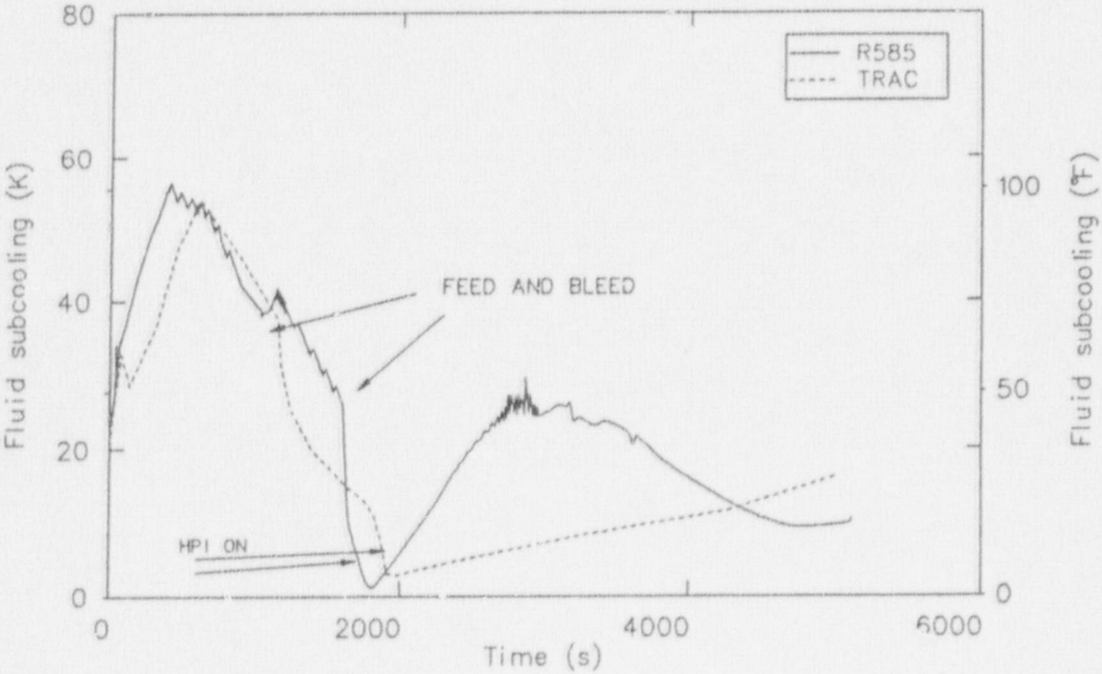

Figure 43. A comparison of calculated fluid subcooling in the A loop hot leg when feed and bleed was initiated near 1500 s.

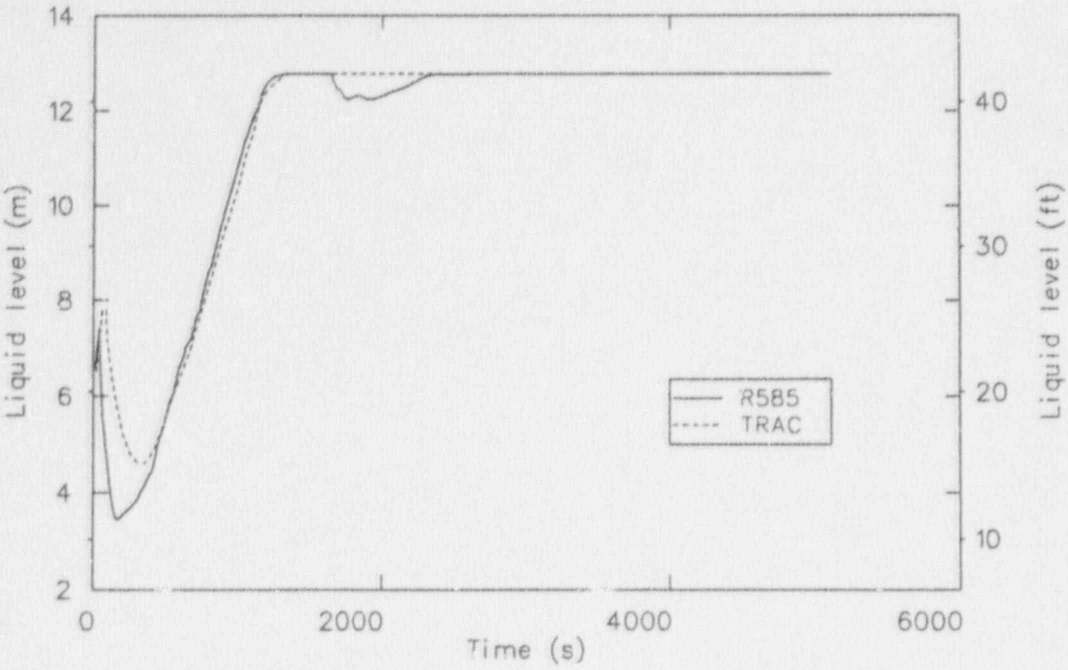

Figure 44. A comparison of calculated collapsed liquid levels in the pressurizer when feed and bleed was initiated near 1500 s.

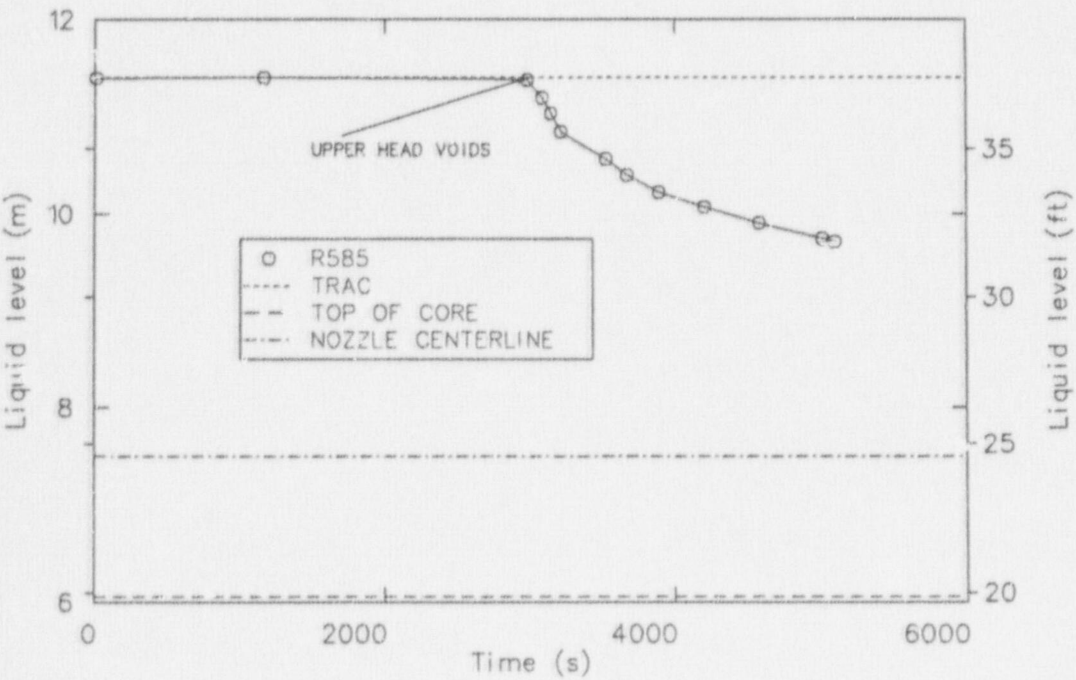

Figure 45. A comparison of calculated collapsed liquid levels in the reactor vessel when feed and bleed was initiated near 1500 s.

to uncertainty in the minimum vessel level for these feed and bleed calculations. However, this uncertainty is not thought to be significant, since the minimum level would be at the bottom of the upper head, well above the top of the core.

The R585 and TRAC calculations showed that if initiated when the hot leg temperature was about 589 K (600°F), feed and bleed would successfully depressurize the RCS and cool the core. No significant voiding occurred in the RCS because the core remained subcooled. HPI flow was initiated relatively early in these calculations because the saturation pressure corresponding to the hot leg temperature at the time of feed and bleed was below the shutoff head of the HPI. Recall that the pressure will quickly drop to the saturation pressure corresponding to the hot leg temperature in the absence of HPI flow. Thus, an extension of the calculations indicates that if feed and bleed is initiated when the hot leg temperature is below 602 K (624°F), corresponding to the saturation temperature of the HPI shutoff head, then the reactor coolant pressure will quickly drop below the HPI shutoff head. If the combined HPI and makeup flows are sufficient to prevent boiling in the core, no significant voiding in the RCS is expected. If the combined flows are not sufficient to prevent boiling in the core, the RCS could pressurize, temporarily shutting off HPI flow and resulting in more extensive voiding of the RCS. Hand calculations indicate that if feed and bleed is initiated before the hot leg temperature reaches 597 K (615°F) and more than 10 minutes have elapsed since reactor trip, the reactor coolant pressure will remain below the HPI shutoff head and no significant voiding in the RCS will occur. The HPI can then cool the core, assuring the success of the feed and bleed operation.

3.5 Calculation Uncertainty

Several potential sources of uncertainty were identified which contributed to the overall uncertainty in the feed and bleed calculations described in this report. The potential sources of uncertainty include the thermal-hydraulic computer code used to make the calculation, the code input model, the initial conditions at the start of the calculation, the boundary conditions applied during the calculation, the code user, and the assumed transient. A review of the calculations and data presented previously in this report helped identify important phenomena relative to the feed and bleed process. These phenomena included the thermal-hydraulic conditions at the start of feed and bleed (principally the RCS temperature) and the boundary conditions during feed and bleed. The important boundary conditions included core power, makeup and HPI flow, and flow through the PORV. The most important measures of the success of feed and bleed were the ability to depressurize and the minimum collapsed liquid level in the reactor vessel. The uncertainties in the calculation of the significant thermal-hydraulic phenomena due to the potential sources of uncertainty were based on several factors, including subjective judgments. Details of this assessment are presented in Appendix C. A summary of the assessment of the uncertainty follows. Unless otherwise stated, the uncertainties apply to the R586 feed and bleed calculation described in Section 3.3.

The uncertainty in the RCS temperature when feed and bleed was initiated in the R586 calculation is estimated to be 5 K (9°F). This uncertainty was caused by uncertainty in the initial and boundary conditions. Table 5 shows the contribution of several different parameters to the uncertainty in the RCS temperature. The largest contributor to this uncertainty was the OTSG dryout time. The uncertainty in the OTSG dryout time was primarily due to uncertainty in the initial liquid inventory in the OTSGs. The stated uncertainty bounds the deviations between the calculated and measured temperatures for the Davis-Besse LOFW transient of June 9, 1985 (see Figures 11 and 12). The stated uncertainty is also representative of the deviation observed between an assessment calculation and data from a feed and bleed test[10] in an experimental facility scaled to a B&W PWR. The stated uncertainty in the RCS temperature corresponds to 11% uncertainty in the calculated heatup rate after OTSG dryout. The corresponding uncertainty in the time required to reach the RCS temperature at which feed and bleed was initiated was about 2 min.

The uncertainties in the R586 calculation due to initial and boundary conditions are also applicable to the R585 calculations. However, the uncertainties in the R585 calculations are larger because of the erroneous

Table 5. Uncertainty in RCS temperature

Parameter	Uncertainty (K)	Uncertainty (°F)
Initial core power	0.8	1.4
Decay heat	1.9	3.5
OTSG dryout time	4.4	8.0
RCP power	0.9	1.6
RCS heat structures	0.8	1.4
Total	5.1	9.1

25% reduction in core power when the RCPs tripped, as discussed in Section 3.4.1. The reduction in core power resulted in a calculated heatup rate that was about 25% too low after RCP trip. This error caused a bias in the R585 calculations in addition to the uncertainty associated with the initial and boundary conditions. For the R585 calculations described in this report, the error caused the calculated RCS temperature to be about 2 K (4°F) too low at the initiation of feed and bleed. The time required to reach the RCS temperature at which feed and bleed was initiated was about 1 min too long. The bias was larger in those R585 calculations, not described in this report, in which feed and bleed was initiated at 37 min. For these calculations, the errors in RCS temperature at the start of feed and bleed and the time to reach this temperature were about 11 K (20°F) and 6 min, respectively.

The uncertainty in the calculated collapsed liquid level in the reactor vessel was estimated to be 1 m (3 ft). This result was based on the results of assessment calculations. In the R586 feed and bleed calculation, an uncertainty was identified relative to the RCP nodalization which could cause the calculated level to be 0.3 m (1 ft) too high. The uncertainty in the level could also be larger because of the upper head modeling for those transients in which feed and bleed was initiated at relatively low RCS temperatures. The uncertainty in the R585 calculation may be larger because the small holes in the plenum cylinder at the hot leg nozzle elevation were not explicitly modeled. However, the uncertainty due to the upper head and upper plenum models should not affect the calculated result that the minimum collapsed liquid level remains above the top of the core.

A simple, quasi-steady volume balance was used to estimate the uncertainty in the calculation of whether or not the RCS should depressurize. The parameters which compress, and thus pressurize, the RCS are the volumetric flow due to makeup, Q_m, and the volumetric expansion due to boiling in the core, Q_c. The volumetric flow through the PORV, Q_p, acts to depressurize the RCS. Figure 46 shows the volumetric flow through the PORV versus quality and the sum of volumetric flows due to makeup and core boiling, $Q_m + Q_c$. The core power was varied parametrically as a function of time after reactor trip from 100% power. The results shown in the figure were obtained at a pressure of 17.2 MPa (2500 psia), which is close to the pressurizer SRV setpoint pressure, and assumed both makeup pumps were available. When the volumetric flow through the PORV exceeds the volumetric flow due to the combination of makeup and core boiling, the RCS will depressurize. Figure 46 shows that the PORV cannot depressurize the RCS for times less than 30 min after reactor trip, regardless of the fluid state at the PORV. However, as time increases, decay

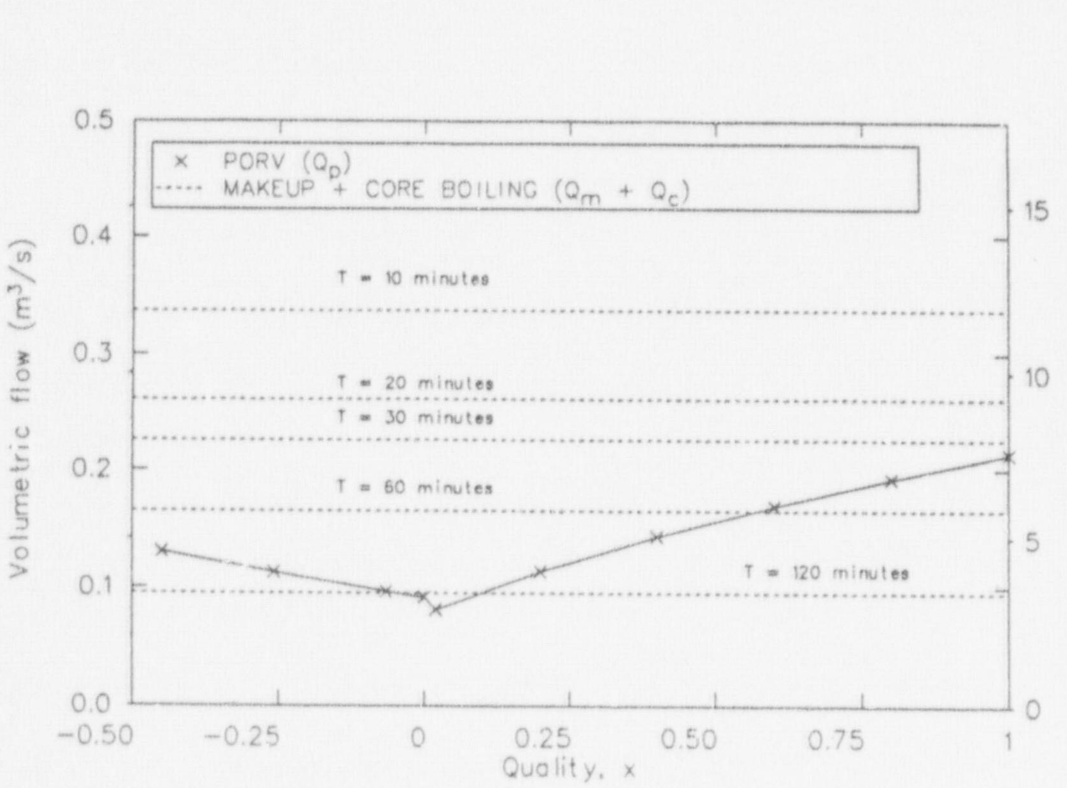

Figure 46. Results of the volume balance for different times after reactor trip.

heat and steam production in the core decrease. At about 37 min after reactor trip, the RCS will depressurize if dry, saturated steam flows through the PORV. The RCS will depressurize with almost any fluid at the PORV at 120 min after reactor trip.

The uncertainty in the boundary conditions of core decay power, PORV flow, and makeup flow does not significantly alter the results of the R586 calculation relative to the depressurization of the RCS. The sensitivity of the calculated depressurization to uncertainty in the feed and bleed boundary conditions is illustrated in Figure 47. The figure shows the best-estimate PORV volumetric flow and the volumetric flow due to makeup plus core boiling at 60 min after reactor trip from Figure 46. This time was selected because it was approximately the time that depressurization occurred in the R586 feed and bleed calculation. The figure also shows the best-estimate PORV flow reduced by 20% and the makeup and core boiling terms resulting from a 5% increase in core decay power and a 10% decrease in makeup flow. These variations correspond to the estimated uncertainties in these parameters.

The effects of the uncertainty in decay power and makeup flow were relatively small. The volume balance indicated that the PORV could still depressurize the RCS at 60 min with either the higher core power or the lower makeup flow. However, a slightly higher quality at the PORV would be required before the depressurization could begin. The uncertainty in the volume balance due to the PORV flow was larger than that due to the uncertainty in the other two parameters. However, the PORV could still depressurize the RCS once the quality at the PORV approached unity. Thus, the uncertainties in the boundary conditions of the R586 feed and bleed calculation do not have a large effect on the calculated ability to depressurize. The calculated depressurization appears valid and would not be expected to vary significantly because of uncertainty in the boundary conditions. The estimated uncertainty in the time of depressurization due to the combined uncertainty in the three boundary conditions is 5 min. Furthermore, sensitivity calculations described in Appendix C indicated that the calculated time of depressurization was not sensitive to hot leg nodalization.

The uncertainty in the depressurization in the R585 calculations was thought to be much larger than described above for those feed and bleed calculations in which the RCPs tripped. As discussed previously, the error in core power when the RCPs tripped significantly affected the depressurization in the R585 calculations.

The above estimates of uncertainty in the important feed and bleed parameters are valid for the transients analyzed based on the assumed initiating event, equipment performance, and operator actions. Significantly

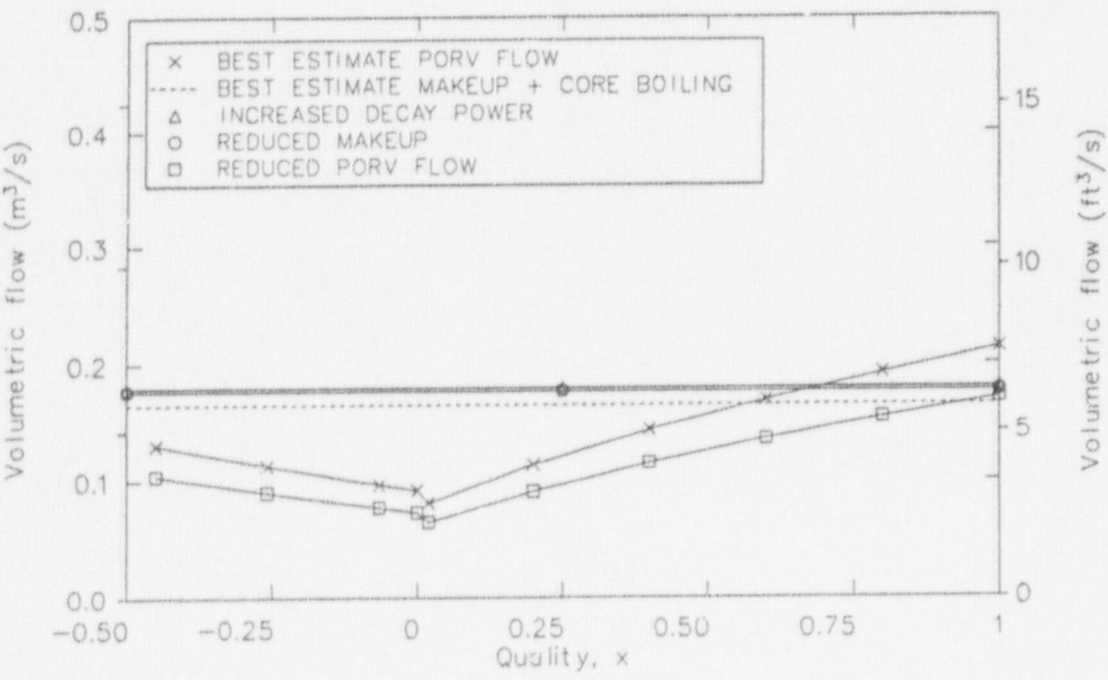

Figure 47. The effect of uncertainty in boundary conditions on the results of the volume balance.

different results could be calculated if different assumptions were made regarding these parameters. Two examples that illustrate the effect of different assumptions on the calculated results are discussed below.

First, in the R585 and R586 feed and bleed calculations, tripping both MFPs resulted in an immediate anticipatory reactor trip. If the anticipatory reactor trip failed or the main feedwater control valves closed while the MFPs continued running, the reactor trip would be delayed until another parameter, such as reactor coolant pressure, reached its trip setpoint. A delay in reactor trip would cause a rapid reduction in steam generator liquid inventory because of the absence of main feedwater. Relatively minor delays in the time of reactor trip thus could result in significantly earlier dryout of the OTSGs and higher reactor coolant temperatures at the initiation of feed and bleed.

Second, in the R585 and R586 calculations, no makeup flow was assumed prior to the initiation of feed and bleed. If a more realistic scenario had been modeled, the operators would have provided maximum makeup flow until pressurizer level was recovered. With maximum makeup, the RCS temperature at the initiation of feed and bleed would have been about 5 K (9°F) lower. Thus, the sensitivity of the calculated results to assumptions regarding operator actions and equipment performance can be as large or larger than the uncertainties described above.

4. CONCLUSIONS

Feed and bleed can be successfully used to cool the core at Davis-Besse in the event that all feedwater is lost. The analysis indicated that if feed and bleed is initiated within 20 min and full makeup flow is available, feed and bleed could successfully depressurize the RCS while cooling the core. The effectiveness of feed and bleed in transients in which the makeup flow was degraded or the initiation of feed and bleed was delayed was not investigated.

The important parameters in a feed and bleed transient are the RCS temperature at the time feed and bleed was initiated, the depressurization during feed and bleed, and the minimum liquid level in the reactor vessel. The RCS temperature at the start of feed and bleed can have a large effect on the transient response, as discussed in Section 3.4.2. Without depressurization, the core could uncover and heatup while the RCS pressure remained above the HPI shutoff head. The minimum liquid level in the reactor vessel determines whether or not the core is adequately cooled.

The R586 calculation of the Davis-Besse LOFW transient was in good qualitative and quantitative agreement with the measured data. As described in Section 3.1, the trends observed in the plant were well represented in the calculation. Prior to reactor trip, the trends of the calculated feedwater flow, main feedwater control valve area, OTSG pressure and level, and RCS response were in excellent agreement with the measured data, indicating that the ICS behavior was well modeled. After reactor trip, the phenomena of OTSG dryout and RCS heatup were also well represented. The maximum deviation between calculated and measured reactor coolant pressure was 0.3 MPa (50 psi). The deviations between calculated and measured reactor coolant temperatures were generally less than 3 K (6°F).

The differences observed between the R586, R585, and TRAC calculations of the Davis-Besse LOFW transient were primarily due to the use of different boundary conditions, including core power, feedwater flow, and pressurizer spray flow. The calculated results were sensitive to the feedwater flow. The effect of variations of the feedwater flow based on its likely uncertainty would probably be as large or larger than the observed differences between the calculations and data. Even though different thermal-hydraulic computer codes and input models were used, the R586, R585, and TRAC calculations were similar and showed trends like those observed in the plant.

The R586 feed and bleed calculation provided a reasonable representation of the transient. The calculation exhibited the phenomena expected. The uncertainty in the RCS temperature at the initiation of feed and bleed was estimated to be 5 K (9°F). The corresponding uncertainty in the time required to reach the RCS temperature at which feed and bleed was initiated was 2 min. The uncertainty in the calculated collapsed liquid level in the reactor vessel was estimated to be 1 m (3 ft).

The uncertainties in the R585 calculations were larger than in the R586 calculation, primarily because of an error in the core decay power following RCP trip. In addition to the uncertainties stated above, the R585 calculations contained a bias in the RCS temperature that ranged from 2 to 11 K (4 to 20°F), depending on the time after RCP trip that feed and bleed began. The corresponding errors in the time to reach a given RCS temperature ranged from 1 to 6 min.

Event timing and RCS pressure response are sensitive to the assumed boundary conditions of core power, PORV flow, and makeup flow. The RCS did not depressurize in the R586 calculation until about 40 min after feed and bleed began. However, as described in Section 3.4, the RCS depressurized immediately when the core power was reduced by 25%. When the PORV flow was reduced by a factor of two, the depressurization was delayed by over an hour. Thus, feed and bleed transients in the plant would also be expected to be sensitive to variations in equipment performance and operator actions.

The feed and bleed transient is significantly affected by the RCS temperature when feed and bleed is initiated. R585 and TRAC calculations showed that when feed and bleed was initiated at a hot leg temperature near 589 K (600°F), the RCS easily depressurized below the shutoff head of the HPI pumps; feed and bleed then successfully cooled the core without significant voiding of the RCS. Hand calculations indicated that no significant voiding of the RCS would occur if feed and bleed was initiated before the hot leg temperature reached 597 K (615°F). If feed and bleed is initiated at a higher RCS temperature, there is the potential for the RCS to pressurize and for significant voiding to occur. In the R586 calculation, the hot leg temperature was about 621 K (657°F) when feed and bleed was initiated. Although feed and bleed was calculated to be successful, the depressurization was delayed and significant voiding occurred in the RCS.

The uncertainty in the boundary conditions of core decay power, PORV flow, and makeup flow did not significantly alter the results of the R586 calculation relative to the ability to depressurize the RCS during feed and bleed. However, hand calculations (see Appendix C) indicated that feed and bleed would not be successful without any makeup flow. In the absence of makeup, the depressurization of the RCS would probably be delayed until after the core heatup began.

The initial and boundary conditions were thought to be the source of most of the uncertainty in the feed and bleed calculations performed. It was recognized, however, that significantly different results could be obtained if different assumptions were made concerning the initiating event, equipment performance, and operator actions during the transient. In particular, the results were thought to be sensitive to assumptions concerning feedwater flow, makeup flow, and reactor trip.

The small holes present in the reactor vessel plenum cylinder, located at the elevation of the hot leg nozzles, should be explicitly modeled. Modeling these holes allows a more accurate representation of the flow paths in the upper plenum, the draining from the hot legs, and the mixture level in the reactor vessel.

The calculated mixture level in the reactor vessel can be sensitive to the modeling of the upper head in certain transients. The upper head may remain subcooled or flash and drain, depending on whether it is modeled as a stagnant or flow-through region. The best modeling technique for the upper head is not known. However, the uncertainty caused by the upper head nodalization is not significant relative to core uncovering for the feed and bleed transients analyzed.

The RCPs should be nodalized such that the correct volume of liquid will remain trapped in the loop seals during a feed and bleed transient. The liquid which remains in the loop seals is not available to cool the core. As described in Appendix C, an uncertainty was identified in the R586 feed and bleed calculation due to RCP nodalization which could cause the calculated reactor vessel liquid level to be about 0.3 m (1 ft) too high.

The results of the R586 feed and bleed calculation were not sensitive to hot leg nodalization. As described in Appendix C, the nodalization of the hot leg affected the timing of events by less than 80 s.

The fission power produced following reactor trip should be modeled. The integrated, posttrip fission power in the R586 calculation of the Davis-Besse LOFW transient was equivalent to 1.5 s of full reactor power. The effect of the posttrip fission power on RCS temperature was about 3 K (6°F), as described in Section 3.2.

5. REFERENCES

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APPENDIX A

R586 MODEL DESCRIPTION

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R586 MODEL DESCRIPTION

A-1. Introduction

The R586 model of Davis-Besse Unit 1 is described in this appendix. The R586 model was developed for use with the RELAP5/MOD2[A-1] computer code and was developed in 1986. The model of the reactor coolant system (RCS) is described in Section A-2. The model of the secondary coolant system is described in Section A-3. Control systems are described in Section A-4. References are presented in Section A-5.

The RELAP5 model of Davis-Besse is based on a model[A-2] of the Oconee-1 pressurized water reactor. The Davis-Besse model is based on the experience gained with the Oconee model, which was developed and used extensively in the pressurized thermal shock program. The main differences between the Oconee and Davis-Besse models are due to geometrical and physical differences between the plants and improvements to the code and modeling techniques since the development of the Oconee model. The modeled geometrical differences include differences in reactor coolant piping due to the raised loop configuration of Davis-Besse; the different condensate, feedwater, and main steam systems; differences in equipment, such as reactor coolant pumps (RCPs), high-pressure injection (HPI), auxiliary feedwater (AFW), etc; and the reduced number of reactor vessel vent valves (four instead of eight). The Davis-Besse vessel model is noded differently than Oconee to allow the use of a crossflow model at the connections between the hot and cold legs and the reactor vessel and the junction between the pressurizer and surge line. The assessment of RELAP5/MOD2[A-3] indicated that improved representation of loop draining could be obtained using the crossflow model at the connections between the loops and reactor vessel. An additional difference between models is that the secondary side of the Davis-Besse once-through steam generator (OTSG) was modeled two-dimensionally to provide a more mechanistic representation of AFW wetting.

A-2. Reactor Coolant System

The model of the RCS is shown schematically in Figures A-1 through A-4. The Davis-Besse plant has a two by four configuration, i.e., two loops, each containing one hot leg and two cold legs. The loops are designated loop A and loop B. Sometimes the loops are also referred to as loop 1 and loop 2, with loop 1 corresponding to loop B. Each loop contains one hot leg, one OTSG, two pump suction legs, two RCPs, and two cold legs (refer to Figures A-1 and A-2). The cold legs in loop A are designated as A1 and A2, corresponding to the RCP number, either 1 or 2, in loop A. The B cold legs are designated similarly. The pressurizer and pressurizer surge line (Figure A-3) are attached to the hot leg in loop A. The RELAP5 vessel model, shown in Figure A-4, represents major components of the vessel, including an inlet annulus, downcomer, lower plenum, core, core bypass, upper plenum, upper head, reactor vessel vent valves, and the control rod guide tube brazements. The major flow path through the downcomer, lower plenum, core, and upper plenum is represented. The minor and leakage flow paths are also represented. These minor flow paths include the core bypass between the core barrel and core former plates, the leakage path between the downcomer and upper plenum around the hot leg nozzles, the leakage path though the control rod guide tube brazements to the upper head, and the small holes in the plenum cylinder at the hot leg elevation. The model of the RCS includes representations of the pressurizer pilot-operated relief valve (PORV) and safety relief valves (SRVs), hot leg high point vent valves (HPVVs), emergency core cooling system (ECCS), makeup and letdown, and pressurizer heaters and spray. The ECCS includes high-pressure injection (HPI), low-pressure injection (LPI), and core flood tanks. LPI and the core flood tanks are connected to the inlet annulus of the reactor vessel. HPI is connected to each of the four cold legs, downstream of the reactor coolant pumps. Heat structures were used to represent heat transfer from and stored energy in the fuel rods, OTSG tubes and tube sheets, loop piping, reactor vessel wall and internals, pressurizer wall, pressurizer surge and spray lines, and pressurizer heaters.

The pressurizer heaters provide a maximum power of 1.329 MW. The pressurizer spray valve is sized to pass 0.012 m^3/s (190 gpm) at normal operating conditions. The PORV is sized to pass 25.2 kg/s (55.5 lbm/s) of saturated steam at 16.10 MPa (2335 psia) and 47.5 kg/s (104.7 lbm/s) of subcooled liquid at 16.46 MPa (2387 psia) and 613 K (644°F). The resulting PORV area was 9.48×10^{-4} m^2 (0.01020 ft^2), with a single-phase liquid discharge coefficient of 0.82 and a two-phase discharge coefficient of 1.0. Each hot leg HPVV is modeled with an area of 1.830×10^{-5} m^2 (0.000197 ft^2), with single-phase and two-phase discharge coefficients of 0.624.

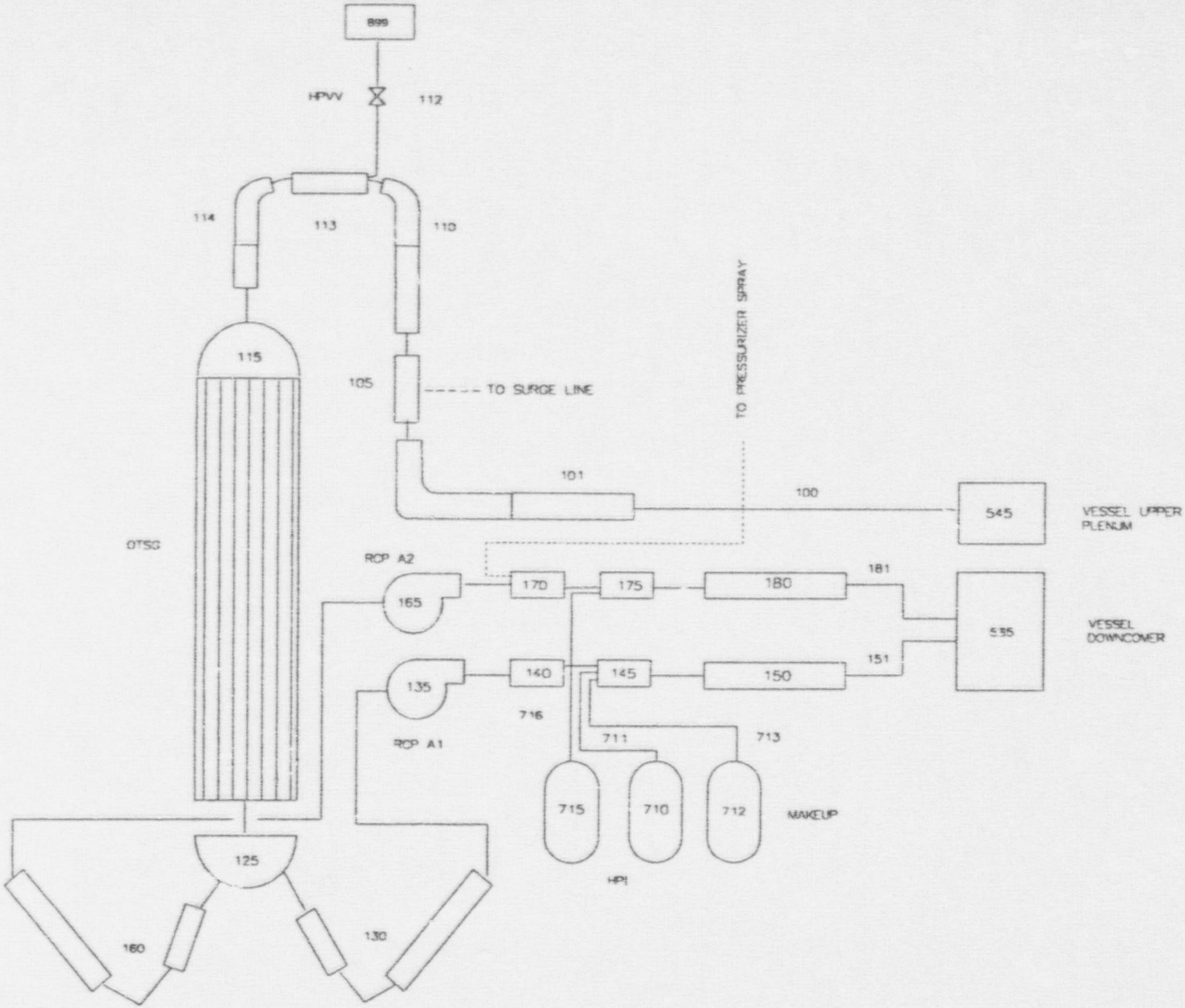

Figure A-1. RELAP5 Davis-Besse model; loop A.

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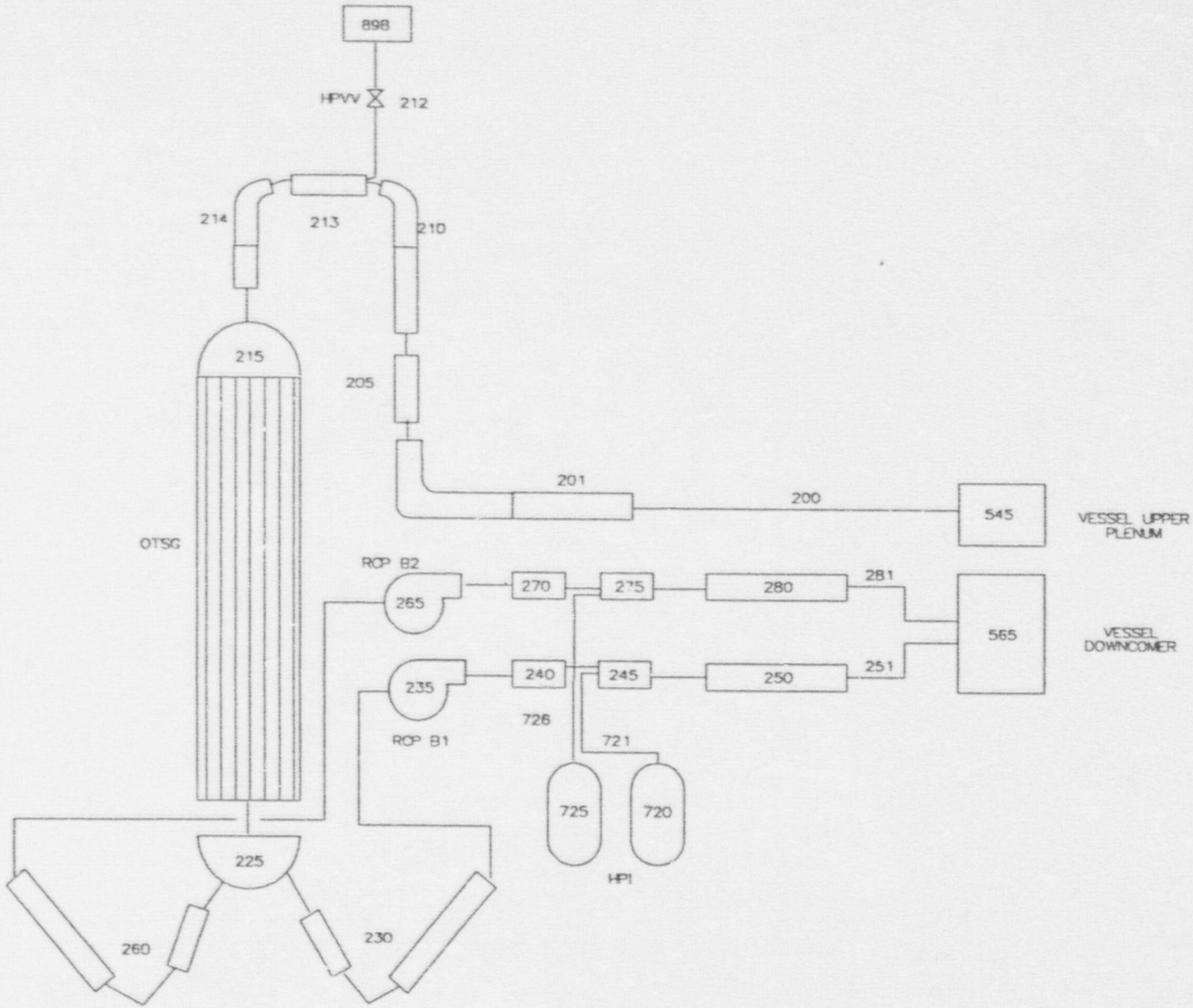

Figure A-2. RELAP5 Davis-Besse model; loop B.

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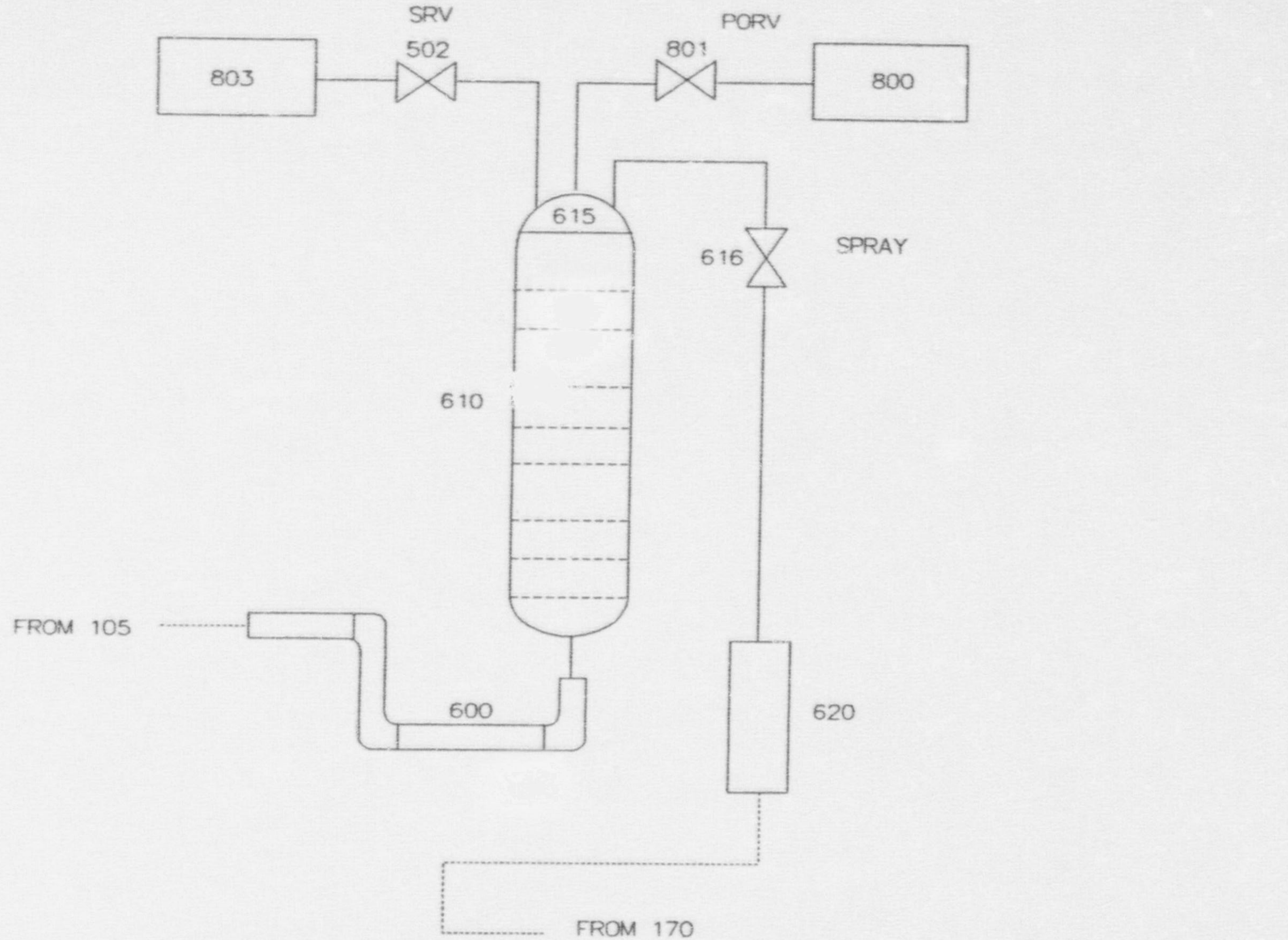

Figure A-3. RELAP5 Davis-Besse model; pressurizer system.

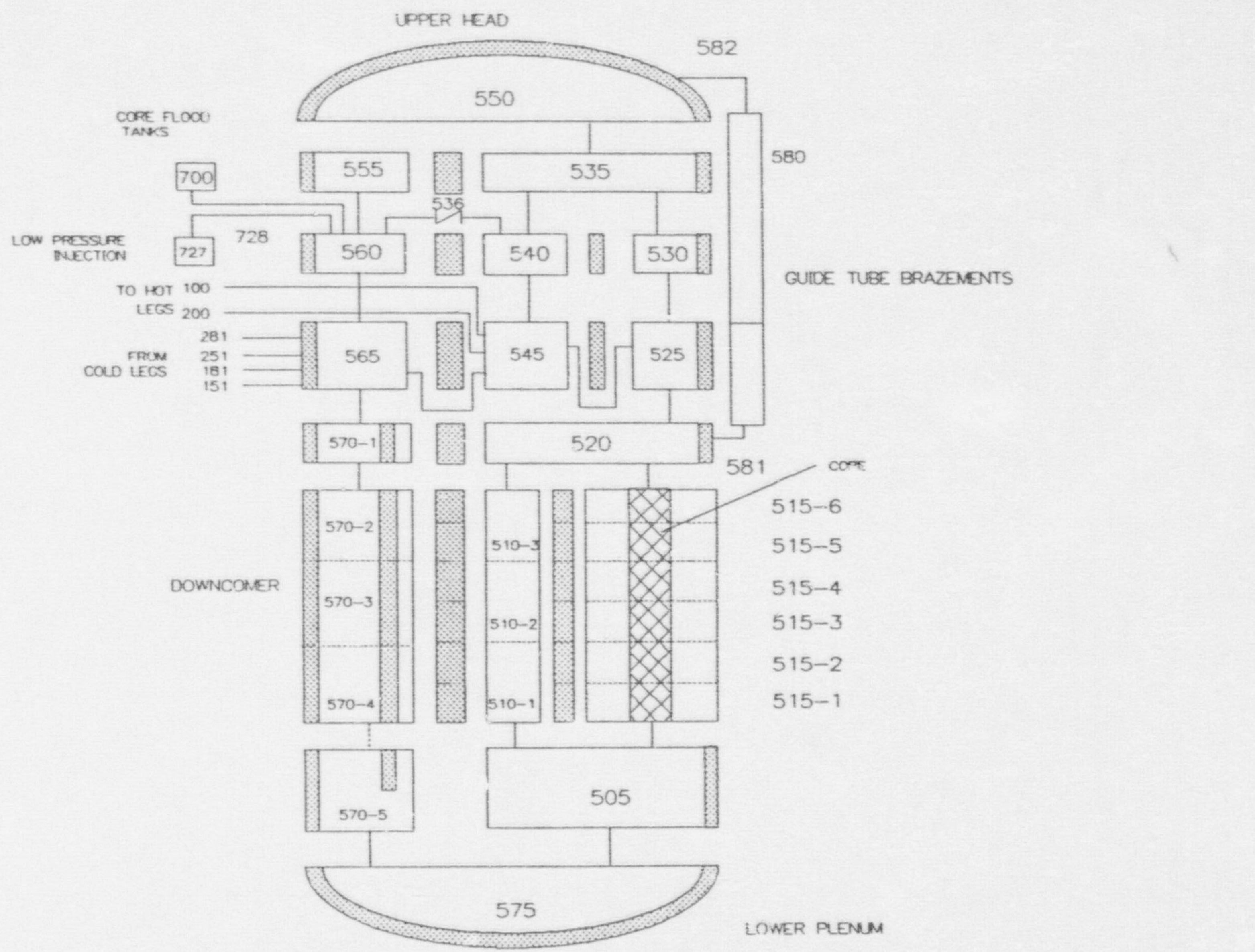

Figure A-4. RELAP5 Davis-Besse model; reactor vessel.

A-3. Secondary Coolant System

The model of the secondary coolant system includes representations of the condensate and main feedwater systems downstream of the deaerator storage tanks, the AFW system, the OTSGs, and the main steam lines. The model of the main feedwater system (see Figure A-5) includes representations of the deaerator storage tanks, turbine-driven booster and main feedwater pumps, the high-pressure feedwater heaters, the main feedwater control and block valves, the startup control valves, the main stop valves, check valves, and connecting piping. Common headers connect the two feedwater trains upstream and downstream of the high-pressure heaters. The deaerator storage tanks are modeled as time-dependent volumes, with pressure and temperature specified as a function of plant load. The power added to the feedwater by the high-pressure heaters is also specified as a function of plant load. The geometry of the high-pressure heaters and the piping upstream of the main feedwater pumps was not available during the development of the model. The piping length upstream of the pumps is assumed, while the geometry of the high-pressure heaters is based on Oconee. Heat structures are used to represent the high-pressure heaters and the piping walls.

The AFW system is illustrated in Figure A-6. The turbine-driven AFW pumps are modeled explicitly to represent the variation in flow to the OTSGs with pump speed. Each pump is normally dedicated to a single OTSG. However, crossover piping downstream of the pumps allows either pump to supply either OTSG. The crossover valve and the admission valves to the OTSGs are controlled by the steam and feed rupture control system (SFRCS).

The Davis-Besse steam generators are once-through and are oriented vertically. Between the outer shell and the heat exchanger tube bundle is a cylindrical baffle, forming a downcomer section. A gap in the baffle allows steam to be drawn from the boiler region into the downcomer to heat the incoming feedwater. After falling through the downcomer, the feedwater enters the tube bundle and flows upwards, vaporizing to saturated steam in the nucleate boiling region. Dry saturated steam is produced in the film boiling region and raised to the exit steam temperature in the superheat region. The steam flow then enters the steam annulus section, which is between the outer shell and the cylindrical baffle and above the feedwater inlet port. The superheated steam then exits the OTSG via the main steam line.

The RELAP5 models of the A and B OTSGs are shown in Figures A-7 and A-8. The major components

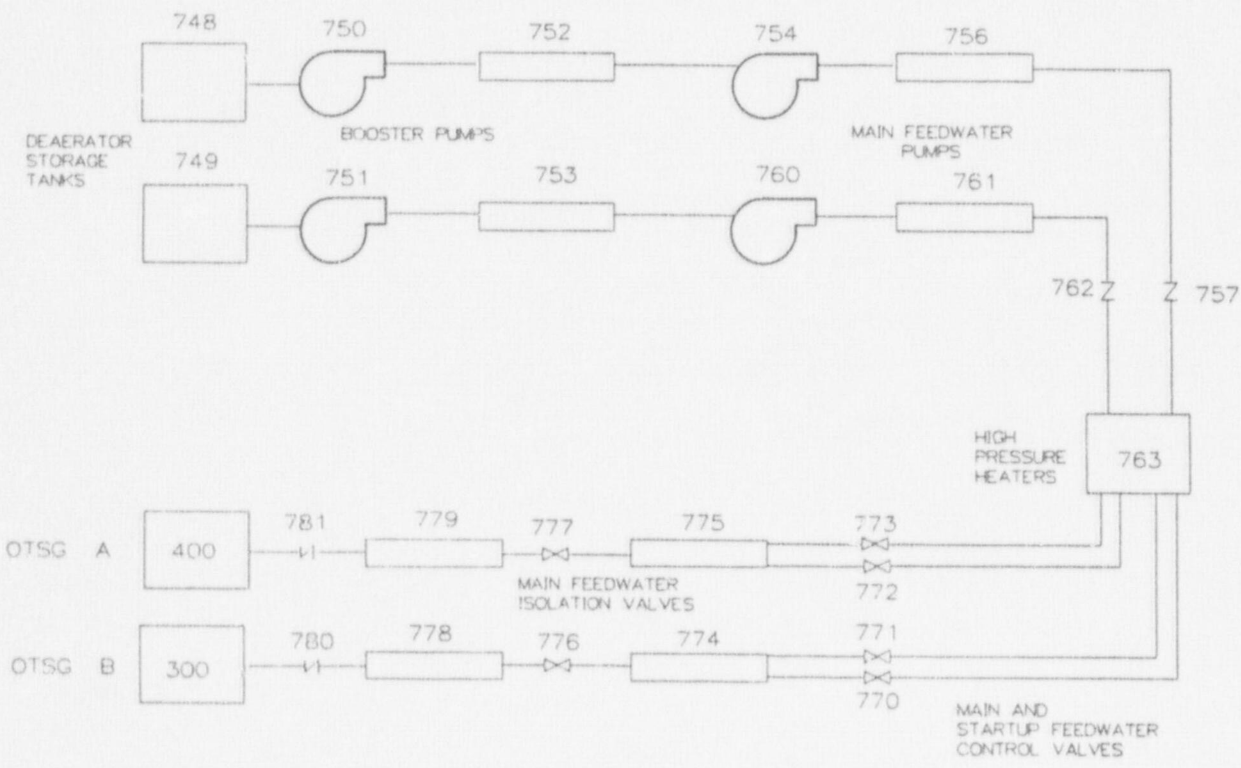

Figure A-5. RELAP5 Davis-Besse model; condensate and feedwater system.

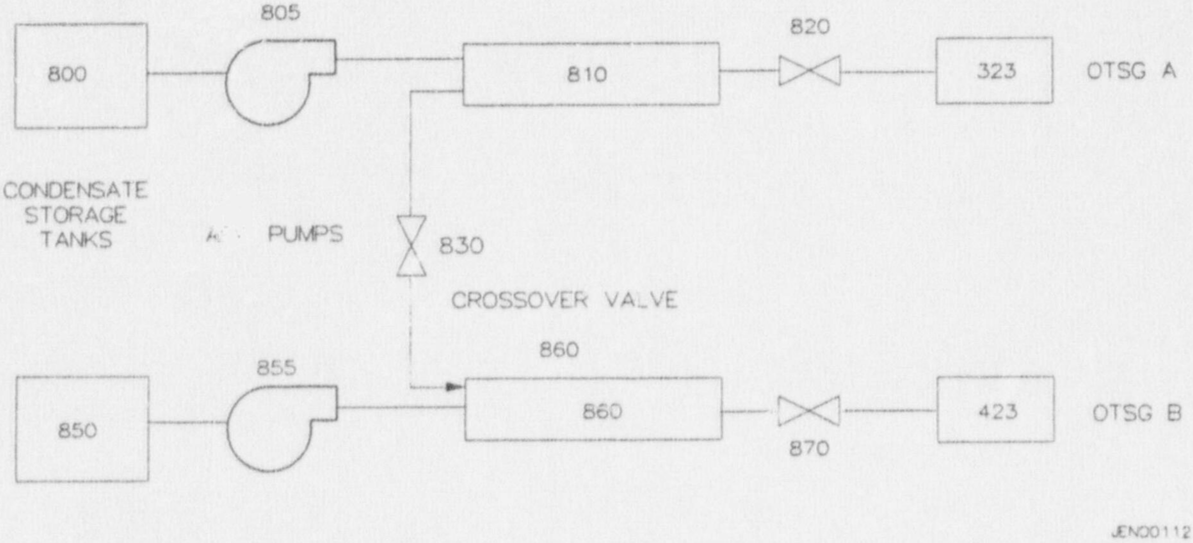

Figure A-6. RELAP5 Davis-Besse model; AFW system.

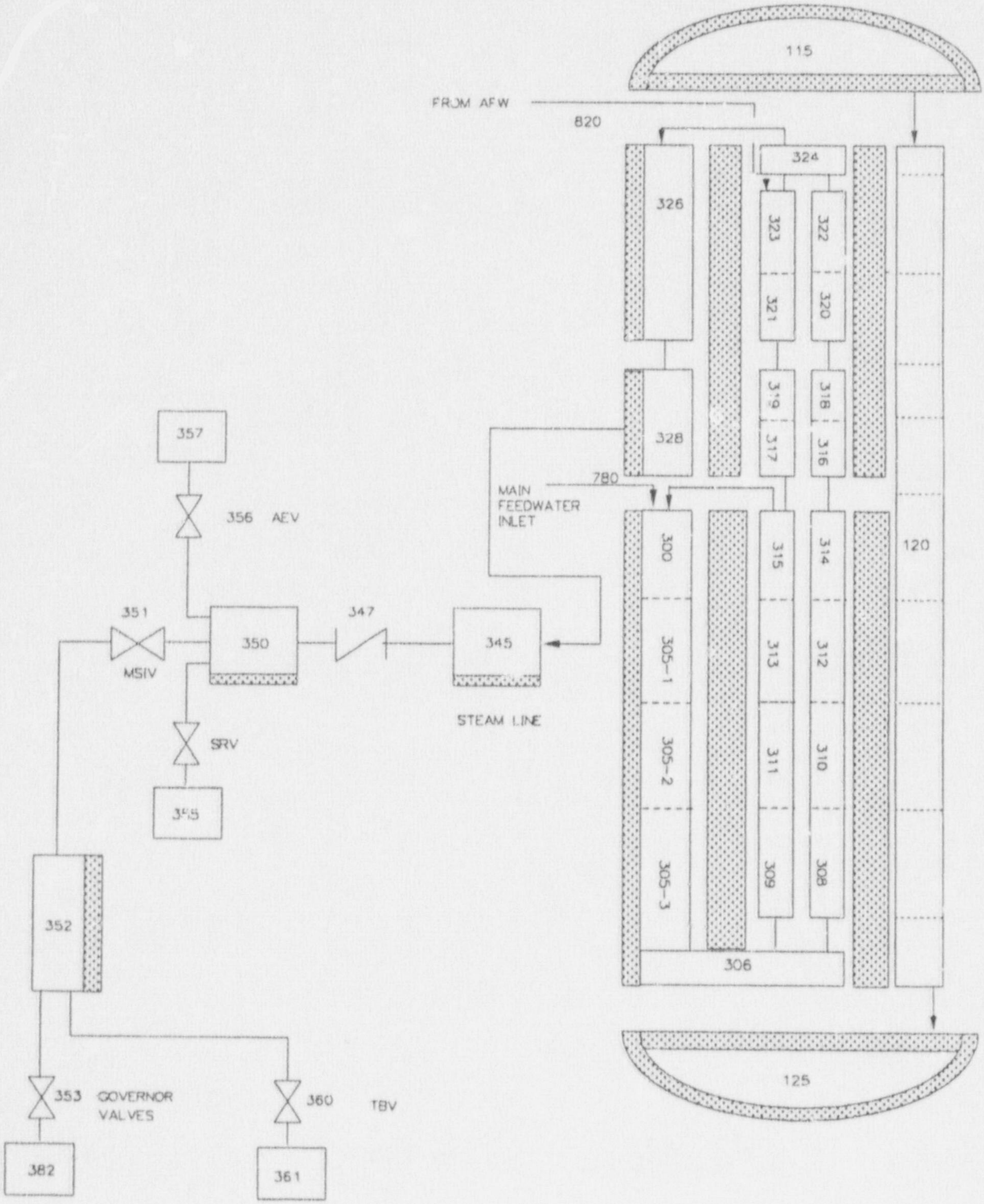

Figure A-7. RELAP5 Davis-Besse model; OTSG A and steam line.

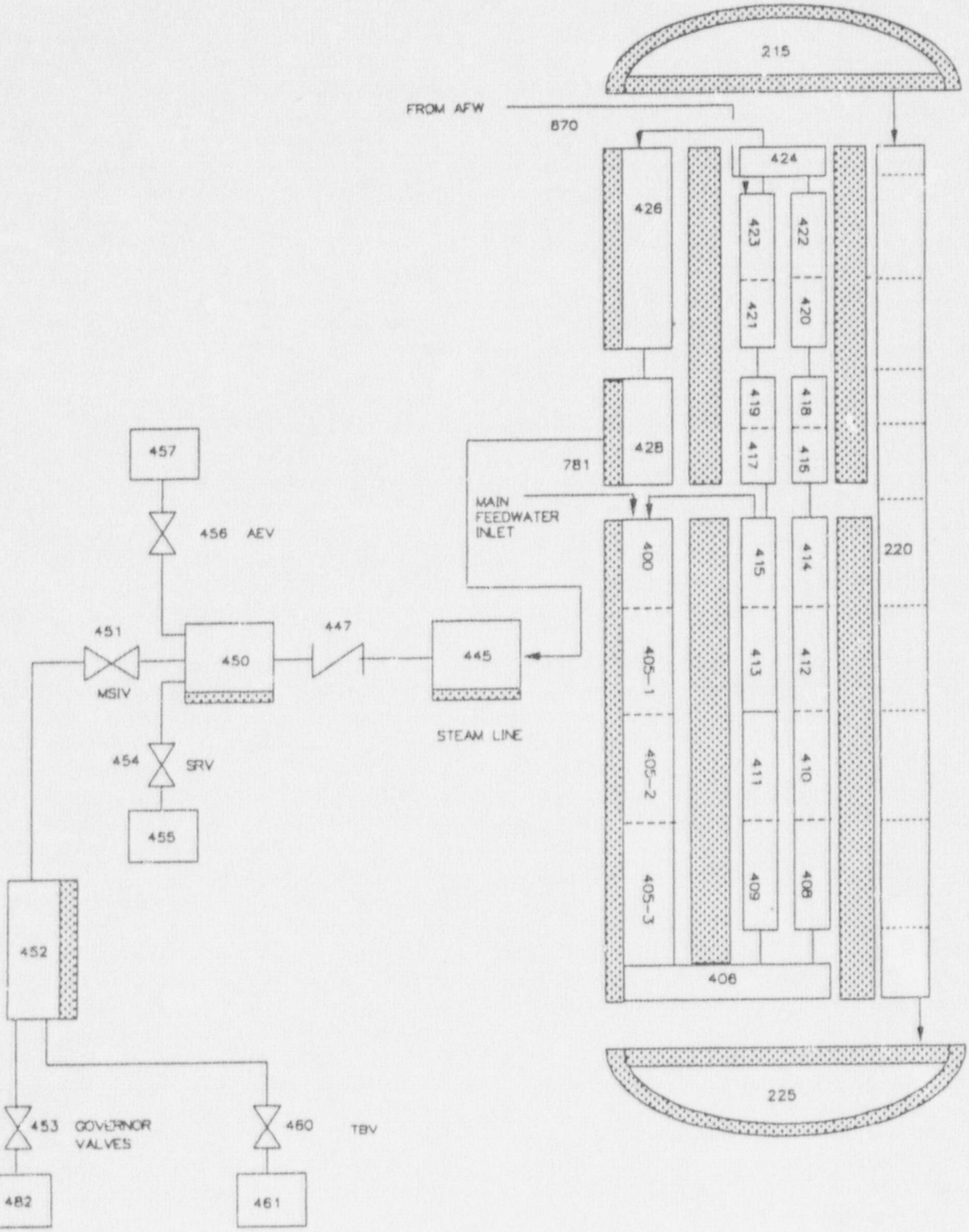

Figure A-8. RELAP5 Davis-Besse model; OTSG B and steam line.

of the OTSG are modeled, including the downcomer, tube bundle region, and steam annulus. AFW enters near the top of the OTSG through eight injection ports which are azimuthally located around the outer edge of the tube bundle. Experiments indicate that the AFW wets only the outer rows of tubes. As the AFW falls, it penetrates radially inwards, but wetting at most 10% of the tubes. The bundle region below the AFW injection ports is divided into two radial regions to approximate the two-dimensional wetting behavior of the AFW. The outer radial region contains 10% of the tubes, while the inner region contains the other 90%. Each radial region is divided axially into eight control volumes. The AFW is connected to the outer radial region. The two regions are connected radially with crossflow junctions, allowing AFW to penetrate radially if calculated by the code. The OTSG tubes are also divided into two separate heat structures, representing 10 and 90% of the tubes. The primary side of the OTSG is modeled with a single channel. Separate-effects calculations performed during the development of the model indicated that two primary channels were not needed for most applications, including natural circulation and small-break transients. Although the multidimensional mixing of the AFW on the secondary side caused significant variations in the behavior of the wetted and unwetted tubes on the primary side, the overall OTSG performance was not significantly different when two primary channels were modeled instead of one. Heat structures are used to represent the stored energy in the secondary shell wall, tube bundle, and cylindrical baffle. The heat transfer hydraulic diameter on the secondary side of the tube bundle is based on the minimum tube-to-tube spacing. This hydraulic diameter improved the thermal performance of the OTSG and was recommended in the assessment of RELAP5/MOD1.[A-4]

The main steam lines models are also shown in Figures A-7 and A-8. The model represents the region from the outlet of the OTSG to the turbine governor valve. The safety relief valves (SRVs), atmospheric exhaust valves (AEVs), turbine bypass valves (TBVs), main steam isolation valves (MSIVs), and check valves are modeled. The turbine stop valve and the turbine governor valve are combined into a single valve in the model. Each AEV and TBV is sized to pass 74.1 kg/s (163.3 lbm/s) and 185.2 kg/s (408.3 lbm/s), respectively, of steam at 6.2 MPa (900 psia). Heat structures are used to represent the steam line piping.

A-4. Control Systems

Many of the plant control systems are represented. These control systems include the integrated control system (ICS), pressurizer pressure control system, anticipatory reactor trip system (ARTS), and steam and feed rupture control system (SFRCS). These control systems are described in greater detail below.

The RELAP5/MOD2 model of the Davis-Besse ICS represents the following subsystems: unit load demand development subsystem, integrated master subsystem, steam generator feedwater control subsystem, and the reactor control subsystem. Figure A-9 is a schematic of the ICS organization and presents an overview of the ICS functions. The borate control subsystem and the non-nuclear instrumentation system are not represented. The ICS model is based on information obtained from Babcock and Wilcox (B&W) and Davis-Besse personnel, plant calibration data, detailed schematics of the subsystems, analog and digital logic drawings, and Bailey Meter Company detailed descriptions of the individual modules.

The ICS modules and relays are modeled individually to provide the greatest amount of flexibility for future analysis requirements. Additional control variables are included in the model to allow the analyst the ability to impose false signals during a calculation. For example, a steam generator level signal can be failed to zero interactively to simulate a failed level transducer. Display parameters and display options available to the operator are also available to the analyst during interactive execution.

The RELAP5/MOD2 kinetics package is not used in the Davis-Besse model. Consequently, reactor control rod positioning is not directly coupled to the reactor power. Instead, the reactor power is controlled by general table reference. Reactor kinetics will be incorporated at a later date, as the need arises.

The pressurizer pressure control system is modeled through the representation of pressurizer heaters and spray. Design data on the pressurizer level control system were not available during the development of the model. Instead, a simple model which controlled the net makeup into the reactor coolant system based on the pressurizer level was developed. The net makeup represented the combination of makeup and letdown, with the net flow added to the A1 cold leg pump discharge. In the plant, letdown is taken from the B1 cold leg pump suction, but the model approximation is thought to be adequate for most applications. The capability to model zero, one, or two makeup pumps and minimum, normal, or maximum letdown, in any combination, was developed.

The model represents the ARTS and SFRCS. Reactor trip is modeled based on high power, high reactor pressure or temperature, power-to-flow ratio, reactor pressure versus temperature, RCP trip, turbine trip, SFRCS actuation, or manual trip. SFRCS is actuated based on low steam pressure, low feedline differential pressure, low or high OTSG level, or reactor

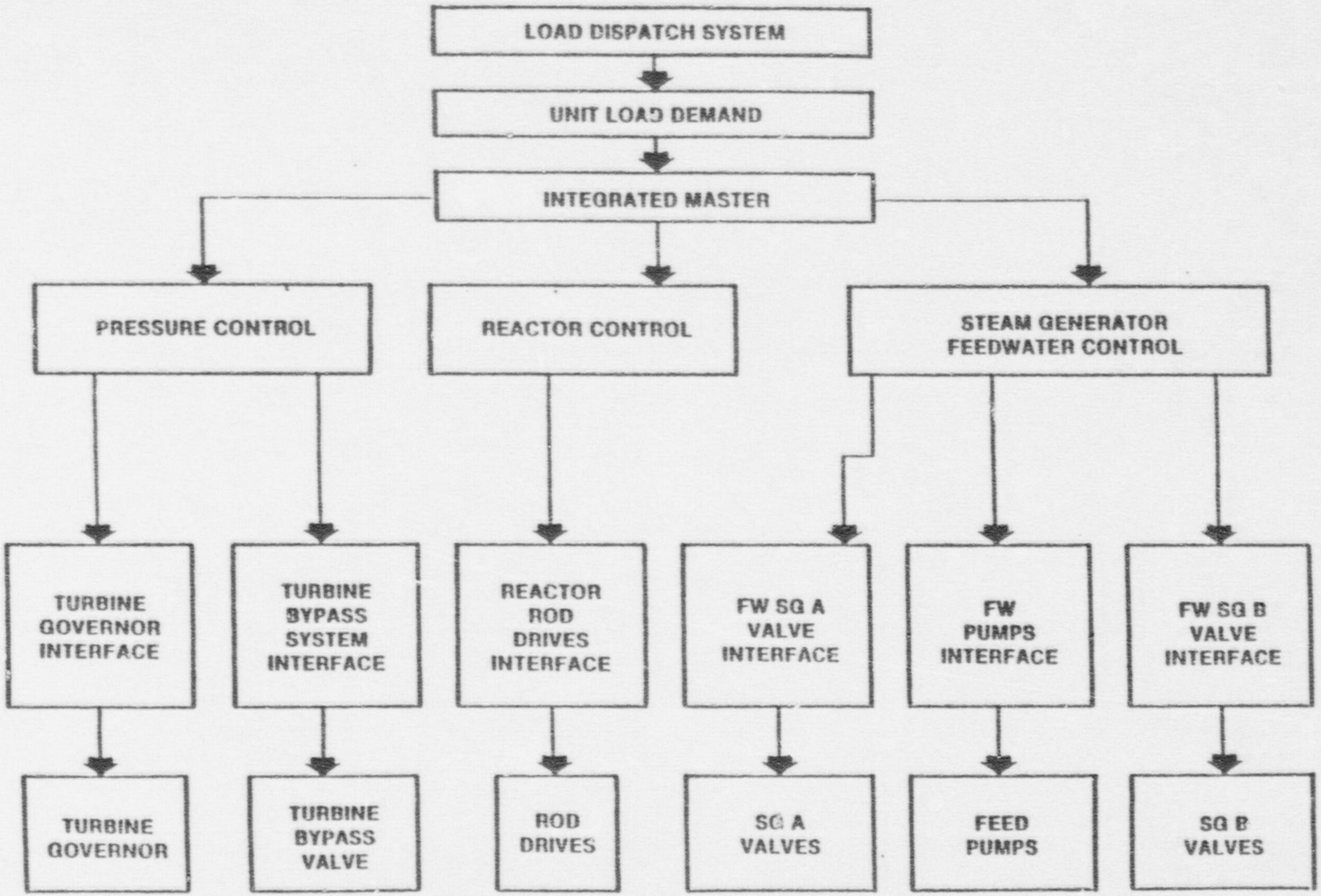

Figure A-9. B&W integrated control system organization.

coolant pump trip. The model determines the correct alignment of AFW based on the the type of SFRCS actuation. In event of a rupture of the steam or feed lines, SFRCS isolates the OTSGs and aligns AFW into the unaffected OTSG.

A-5. References

A-1. V. H. Ransom et al., *RELAP5/MOD2 Code Manual, Volumes 1 and 2*, NUREG/CR-4312, EGG-2396, August 1985.

A-2. C. D. Fletcher et al., *RELAP5 Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor*, NUREG/CR-3761, EGG-2310, June 1984.

A-3. P. D. Wheatley et al., *RELAP5/MOD2 Code Assessment at the Idaho National Engineering Laboratory*, NUREG/CR-4454, EGG-2428, March 1986.

A-4. S. L. Thompson and L.N. Kmetyk, *RELAP5 Assessment: LOFT Large Break L2-5*, NUREG/CR-3608, SAND83-2549, February 1984.

APPENDIX B
COMPUTER RUN TIME STATISTICS

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COMPUTER RUN TIME STATISTICS

Table B-1 presents a summary of the computer run time statistics of the RELAP5 calculations performed at the Idaho National Engineering Laboratory (INEL) during this task. These calculations were described in Sections 3.1 and 3.3 of the main body of this report and represented the loss-of-feedwater (LOFW) transient that occurred at Davis-Besse on June 9, 1985, and a total LOFW followed by feed and bleed. The computer used to perform the calculations was the Cyber 176 at the INEL. The calculations were performed with version 36.04 of the RELAP5/MOD2 computer code.

The feed and bleed calculation executed more quickly than the Davis-Besse LOFW transient calculation. The code was able to take larger time steps after the reactor coolant pumps were tripped in the feed and bleed calculation. The code performed reliably for both calculations. No code execution failures were encountered in either calculation.

Table B-1. Computer run time statistics

Parameter	LOFW Calculation	Feed and Bleed Calculation
Number of volumes (C)	201	201
Number of heat transfer surfaces	291	291
Transient time, s (RT)	765	5800
Total CPU time, s (CPU)	3469	17290
Number of time steps (DT)	10263	50937
CPU/RT	4.53	2.98
(CPU x 10)/(RT x C)	0.23	0.15
(CPU x 1.E6)/(RT x C x DT)	2.20	0.29
(CPU x 1000)/(C x DT)	1.68	1.69

APPENDIX C
CALCULATION UNCERTAINTY

APPENDIX C

CALCULATION UNCERTAINTY

C-1. Introduction

Several potential sources of uncertainty were identified which contributed to the overall uncertainty in the feed and bleed calculations described in this report. The potential sources of uncertainty include the thermal-hydraulic computer code used to make the calculation, the code input model, the initial conditions at the start of the calculation, the boundary conditions applied during the calculation, the code user, and the assumed transient. A review of the calculations and data presented in the main body of this report helped identify important phenomena relative to the feed and bleed process. These phenomena included the thermal-hydraulic conditions at the start of feed and bleed, principally the reactor coolant system (RCS) temperature, and the boundary conditions during feed and bleed. The important boundary conditions during feed and bleed included core power, makeup and high-pressure injection (HPI) flow, and flow through the pilot-operated relief valve (PORV). The most important measures of the success of feed and bleed were the ability to depressurize and the minimum collapsed liquid level in the reactor vessel.

The uncertainties in the calculation of the significant thermal-hydraulic phenomena due to the potential sources of uncertainty were assessed, quantitatively when possible, as described in Section C-2. The estimated uncertainties in the important parameters are summarized in Section C-3. References are presented in Section C-4. Unless otherwise stated, the uncertainties apply to the R586 calculation performed at the Idaho National Engineering Laboratory (INEL), as described in Section 3.3 of the main body of this report.

C-2. Evaluation of Potential Sources of Uncertainty

The potential sources of uncertainty in the feed and bleed calculations include the thermal-hydraulic computer code, the code input model, the initial conditions and boundary conditions, the code user, and the assumed transient. The evaluation of the uncertainty in the feed and bleed calculations due to each of these potential sources of uncertainty appears in Sections C-2.1 through C-2.5.

C-2.1 Thermal-Hydraulic Computer Code. The RELAP5/MOD2 computer code[C-1] was used to perform the R586 feed and bleed calculation. In general, the uncertainty of a computer code for calculating feed and bleed transients should ideally be determined by an assessment program which involves comparisons of calculations and data from a wide range of scale and test parameters. Unfortunately, however, only limited data and assessment calculations exist for feed and bleed in Babcock and Wilcox (B&W) geometry. One directly applicable assessment calculation, utilizing feed and bleed data from Once-Through Integral System Test 220899,[C-2] indicated that RELAP5/MOD2 provided an excellent representation of the observed trends. The deviations between calculated and measured RCS pressures, RCS temperatures, and reactor vessel liquid levels were generally less than 0.7 MPa (100 psi), 5 K (9°F), and 1 m (3 ft), respectively. Although this test was not scaled to the Davis-Besse feed and bleed transient described in Section 3.3, the assessment results described above are thought to be representative.

Additional data comparisons have been performed which provide an assessment of the ability of RELAP5/MOD2 to calculate some of the important phenomena occurring in a feed and bleed transient. Feed and bleed transients exhibit relatively simple thermal-hydraulic processes, involving dryout of the once-through steam generators (OTSGs), heatup of the RCS, initiation of feed and bleed, draining of the RCS down to a minimum level, and refill. The ability of the code to predict the phenomena of OTSG dryout and RCS heatup was demonstrated in the calculation of the Davis-Besse LOFW transient of June 9, 1985, as described in Section 3.1 of the main body of this report. The ability of the code to predict RCS draining and refill in pressurized water reactors (PWRs) with U-tube steam generators has been demonstrated in assessment calculations[C-3] of small-break loss-of-coolant accidents. Calculated liquid levels were generally within 1 m (3 ft) of the corresponding data. Assessment calculations of feed and bleed experiments applicable to U-tube steam generator PWRs have also been performed.[C-4,5] These feed and bleed calculations, although not directly applicable because early versions of the code were used, indicated that RELAP5 was generally able to predict the trends observed in the experiments. Consequently, RELAP5/MOD2 should adequately represent feed and bleed transients in B&W PWRs. The uncertainty inherent in the code for calculating feed and bleed transients, although difficult to quantify, is not thought to be large compared with other sources of uncertainty.

Another method for estimating the uncertainty of a computer code is to compare the calculated results from one code with those from another code. The variation between the calculated results provides an indication of code uncertainty. Sections 3.2 and 3.4.2 present comparisons of calculated results from the RELAP5/MOD2 and TRAC-PF1/MOD1[C-6] computer codes. The comparisons showed that the codes obtained similar macroscopic results in that the feed and bleed operations successfully cooled the core. However, the codes calculated significantly different trends, particularly with respect to RCS pressure. The differences between the calculations were attributed to errors in the boundary conditions. The differences in the boundary conditions were thought to overwhelm differences between the codes. Consequently, the differences in the RELAP5 and TRAC results were not indicative of the uncertainty in either code. Both codes are thought to have the capability to adequately represent feed and bleed transients.

C-2.2. Input Model. The uncertainty of the calculations due to the input model primarily arises from two different concerns. These concerns include the accuracy with which the model reflects the geometry of the plant and the adequacy of the nodalization for the feed and bleed transient.

In general, the R586 RELAP5/MOD2 model is believed to accurately represent Davis-Besse. The detailed information used in the development of the model and the quality-assurance procedures yield a high degree of confidence that the model adequately represents Davis-Besse. The agreement between data from the Davis-Besse LOFW transient of June 9, 1985, and the corresponding calculation (see Section 3.1) further increases the confidence in the basic geometry, the response of the integrated control system, and the applied boundary conditions. Thus, the uncertainty in the feed and bleed calculations due to uncertainty in the basic input model is thought to be negligible compared to other sources of uncertainty. However, even with the overall confidence in the model, the model does have limitations (see Appendix A), primarily related to the lack of information during the development of the model. The uncertainty in the R585 calculations performed by the Nuclear Regulatory Commissions's Office of Nuclear Reactor Regulation (NRR) due to the geometry of the input model is also thought to be small. As described in Sections 3.2 and 3.4.1, the R585 calculations were generally in good agreement with the R586 calculations except for differences associated with boundary conditions of core power, pressurizer spray, and feedwater flow.

Part of the uncertainty associated with the input model is caused by the nodalization used. The nodalization of several different components has the potential to cause uncertainty in feed and bleed calculations. These components include the OTSGs, reactor vessel upper plenum and upper head, heat structures, hot legs, and reactor coolant pumps (RCPs) and RCP suction legs. A discussion of the possible uncertainty due to the nodalization of these components follows.

The OTSG nodalization is not thought to significantly contribute to the uncertainty of the R586 feed and bleed calculation. The most important OTSG parameter is the initial liquid inventory. Since the OTSGs dry out relatively quickly, and since the total liquid inventory must boil away eventually, the OTSG nodalization is not significant for feed and bleed calculations. Of course, the OTSG nodalization might significantly affect those calculations in which the OTSGs actively remove heat throughout the transient.

Calculated reactor vessel liquid levels can be sensitive to the nodalization of the upper plenum and upper head. As shown in Figure 35 of the main body of this report, the R585 model predicted liquid levels that were about 1.5 m (5 ft) higher than the R586 model. About half of this difference was attributed to the different upper plenum models used in the calculations. The R586 model explicitly represented the small holes in the plenum cylinder at the hot leg nozzle elevation, while the R585 model did not. Modeling these holes allowed a more accurate representation of the flow paths in the upper plenum, the draining from the hot legs, and the mixture level in the vessel. The upper plenum model used in the R585 calculations thus caused the calculated liquid level to be about 1 m (3 ft) too high for certain transients. Section 3.4.2 explained that the reactor vessel liquid levels could be significantly different, depending on whether the upper head was modeled as a flow-through or a stagnant region. It is not known which model generally best represents the upper head. Although the uncertainty in level caused by the upper head model can be as large as the upper head height, about 2 m (6 ft), the uncertainty appears only for those transients in which feed and bleed was initiated at relatively low RCS temperatures. Since the minimum liquid level remains at or above the bottom of the upper head for these transients, the uncertainty is not significant relative to cooling the core during feed and bleed.

Some of the uncertainty in a feed and bleed calculation is due to the model of the RCS heat structures. The RCS heats up after the OTSGs dry out. The coolant heatup rate is slowed by the heat structures, which absorb energy and heat up along with the coolant. For example, after the OTSGs dried out in the R586 calculation of the Davis-Besse LOFW transient, the heat structures absorbed energy at a rate equivalent to 20% of the core decay power. The important parameters in the heatup of the heat structures are mass and specific heat capacity. If the heat structures are neglected, the calculated heatup rate will be too large. Since all the major heat structures were represented in the R586 model, the uncertainty in the heatup rate due to the heat structures was thought to be relatively small.

The R586 feed and bleed calculation showed that the final RCS depressurization was coupled to the increase in void fraction at the PORV. This increase in void fraction occurred when the mixture level dropped below the pressurizer surge line, allowing steam to pass out through the PORV. The hot leg nodalization affects the hot leg mixture level and the void fraction passed to the surge line. Consequently, a sensitivity calculation was performed to investigate the effect of hot leg nodalization. The sensitivity calculation was restarted from the base calculation, described in Section 3.3, at 1600 s. There was no significant voiding in the hot legs at this time. The A loop hot leg was renoded by reducing the length of the volume connected to the surge line from 6 m (20 ft) to 1.5 m (5 ft) while keeping the total number of volumes constant. The calculated results were insensitive to the hot leg nodalization. The nodalization had only a small effect on hot leg liquid level (Figure C-1) and RCS pressure (Figure C-2). The change in hot leg nodalization affected the timing of events by less than 80 s. Thus, the uncertainty in the calculated feed and bleed results due to hot leg nodalization is insignificant.

The purpose of the feed and bleed operation is to cool the core in the absence of a secondary heat sink. Keeping the core covered with liquid assures that the core will be cooled. The core is not in danger of uncovering when the mixture level is above the pressurizer surge line. However, in the R586 feed and bleed calculation, the RCS was losing mass and could not depressurize with liquid flowing through the PORV. After the hot leg mixture level dropped below the surge line and steam passed through the PORV, the RCS depressurized but continued to lose mass, and the mixture level approached the top of the core. The mixture level decreased until the makeup flow exceeded the flow out the PORV. For the feed and bleed operation to be successful, the reactor vessel mixture level should remain above the top of the core until the RCS is depressurized far enough so that the makeup and HPI systems can maintain liquid inventory.

The minimum liquid level in the vessel depends on the minimum liquid levels reached in the OTSGs and the loop seals of the RCP suction piping. The Davis-Besse RCPs contain a weir which determines the minimum level in the loop seals. The liquid in the RCPs and OTSGs above the weir should drain into the vessel

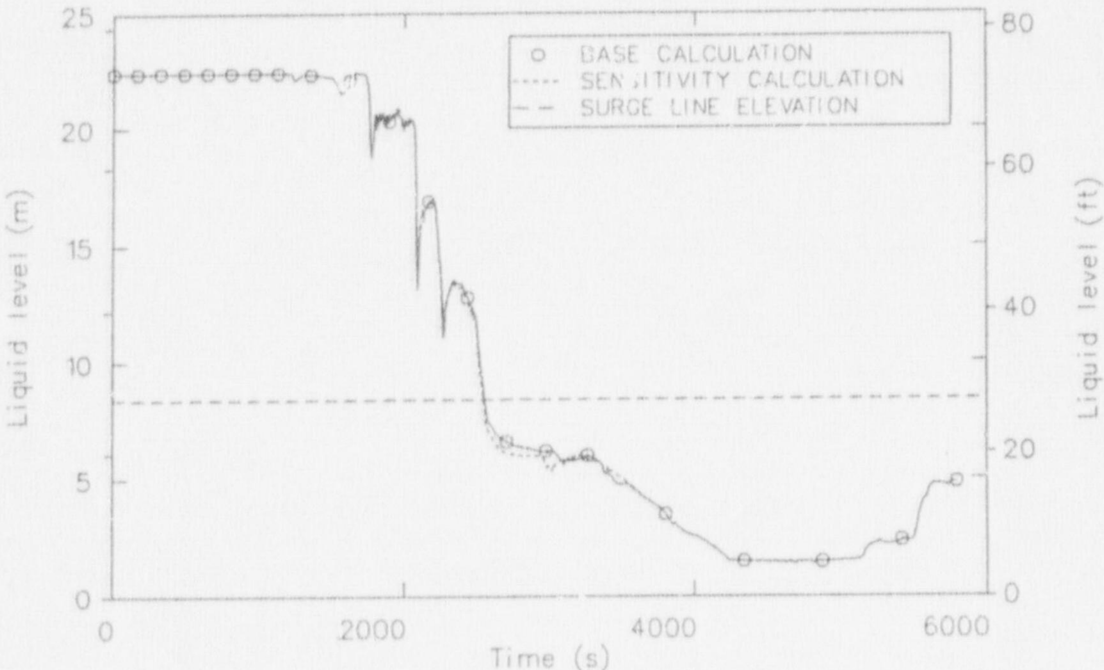

Figure C-1. The effect of hot leg nodalization on collapsed liquid level in the A loop hot leg.

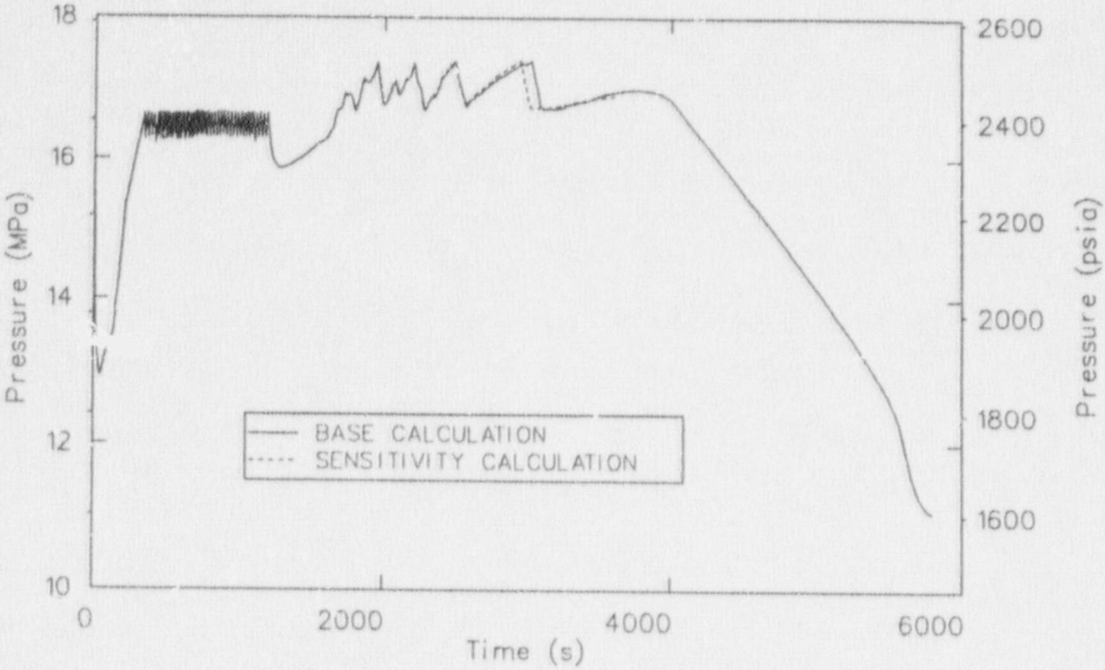

Figure C-2. The effect of hot leg nodalization on RCS pressure.

and cool the core. The liquid below the weir should remain trapped in the loop seal and be unavailable to cool the core. The elevation of the weir was not available during the development of the model and thus was not incorporated into the model. The RELAP5 model has the potential to drain the loop seals down to the bottom of the RCPs, allowing too much water to drain into the vessel. Thus, the calculated minimum vessel liquid level could be too high when the level approaches the top of the core. The uncertainty in the minimum reactor vessel liquid level due to the RCP nodalization was estimated to be about 0.3 m (1 ft) in the R586 feed and bleed calculation.

C-2.3 Initial and Boundary Conditions. The RCS temperature at the start of feed and bleed can have a large effect on the transient response, as discussed in Section 3.4.2. Operator guidelines may also direct that feed and bleed cooling be started when the hot leg reaches a certain temperature. Thus, the RCS temperature at the start of feed and bleed is an important parameter. The ability to depressurize is also an important parameter because without depressurization the core could uncover and heat up while the RCS pressure remains at the open setpoint of the pressurizer safety relief valves (SRVs). The effect of uncertainty in the initial and boundary conditions on these two important parameters was determined in the R586 feed and bleed calculation as follows. First, the initial and boundary conditions which had the potential to affect the important parameters were identified. Second, the uncertainty in each of these initial and boundary conditions was estimated. Third, the uncertainty in the initial and boundary conditions was propagated linearly to determine the total uncertainty in the important parameters. The method for determining the uncertainty and the results obtained are described in more detail below.

The initial and boundary conditions which had the potential to affect the RCS temperature and the depressurization are listed in Table C-1, along with estimates of uncertainty. The uncertainties represent estimates of two standard deviations. The probability was thought to be about 95% that the true value of the initial or boundary condition was within the stated uncertainty of the calculated or input value. The estimates of uncertainty were generally subjective and were based on engineering judgment. The uncertainty in the core power was divided into three components: those associated with the initial power, the decay power (including actinides), and the fission power produced after reactor trip. The uncertainty in the fission power was thought to be larger than the uncertainty in the other two components. The uncertainty in the initial OTSG liquid mass was based on a calculation of the variation in mass due to tube fouling. The uncertainty in the PORV flow was probably less than 10% for single-phase thermodynamic states upstream of the valve because of the available test data. The estimated uncertainty was increased to 20% to

Table C-1. Uncertainty in initial and boundary conditions

Parameter	Uncertainty (%)
Core power	
Initial power	2
Post-trip fission power	35
Decay heat	5
PORV flow	20
Makeup flow	10
Initial OTSG liquid mass	14
Post-trip feedwater flow	100
Initial stored energy in fuel	10
RCP power	10
RCS heat structures	10

account for the effects of two-phase flow through the PORV.

The total uncertainty in the important parameters (the RCS temperature and the ability to depressurize) was obtained by combining the effects due to the uncertainty in the individual contributors (the initial and boundary conditions). The effect of a variation in each individual contributor by its uncertainty on an important parameter was estimated with a hand calculation. The change in the important parameter from the code-calculated result was assumed to be a linear function of the individual contributors. The individual contributors were assumed to be independent, normally distributed random variables. The total uncertainty in the code-calculated result was then determined with a standard statistical formula: the square root of the sum of the squares of the uncertainty due to each individual contributor. Although the above method was not entirely rigorous or statistically justified, the method is thought to provide a reasonable, but rough, estimate of the actual uncertainty.

The RCS temperature at the initiation of feed and bleed was primarily determined by the time of OTSG dryout and the heatup rate after dryout. The analysis of the R586 feed and bleed calculation indicated that the parameters which primarily determined OTSG dryout were the initial liquid mass in the OTSGs, the amount of feedwater delivered to the OTSGs after reactor trip, the core and RCP power, the stored energy in the fuel, and the OTSG pressure and temperature response as controlled by the atmospheric exhaust valve (AEV). The effect of the AEV was neglected in the estimate of the uncertainty because, although it affects the OTSG dryout time, its ultimate effect on RCS temperature was minimal. The uncertainty in the calculated OTSG dryout time due to uncertainty in the initial and boundary conditions was calculated as described above. The OTSG dryout time varied between about 50 and 240 s, while the OTSGs dried out at 130 s in the R586 calculation. Thus, the uncertainty in the calculated OTSG dryout time was about 100 s. The major contributor to the uncertainty in the OTSG dryout time was the initial OTSG liquid mass.

The analysis of the R586 calculation indicated that the parameters which most significantly affected the RCS temperature at 20 min, the time when feed and bleed was initiated, were the time of OTSG dryout and the net power added to the reactor coolant after OTSG dryout. The net power included core decay power, RCP power, and the power absorbed by heat structures during the RCS heatup. The uncertainty in the calculated RCS temperature at 20 min due to the uncertainty in each individual contributor (Table C-1) is shown in Table C-2. The largest contributions to the uncertainty in RCS temperature were associated with the OTSG dryout time and the core decay heat. The combined (total) uncertainty in the calculated RCS temperature due to the individual uncertainties shown in Table C-2 was 5 K (9°F). The corresponding uncertainty in the calculated heatup rate was 11%. Based on the average heatup rate, the uncertainty in the time at which a given temperature was reached was about 2 min. The above uncertainties are also thought to be applicable to the R585 calculations. As described previously, the R585 and R586 calculations were generally similar prior to the initiation of feed and bleed.

The depressurization of the RCS is an important parameter in feed and bleed. The R586 feed and bleed

Table C-2. Uncertainty in RCS temperature

Parameter	Uncertainty (K)	Uncertainty (°F)
Initial core power	0.8	1.4
Decay heat	1.9	3.5
OTSG dryout time	4.4	8.0
RCP power	0.9	1.6
RCS heat structures	0.8	1.4
Total	5.1	9.1

calculation indicated that the RCS would depressurize when the void fraction at the PORV reached a high enough value. Hand calculations were performed to estimate the uncertainty in the calculated depressurization. The analysis of the feed and bleed calculations described in Section 3.4 indicated that the key boundary conditions affecting the depressurization were core power, PORV flow, and makeup flow.

A simple, quasi-steady volume balance was used to estimate the uncertainty in the calculation of whether or not the RCS should depressurize. The parameters which act to compress, and thus pressurize, the RCS include makeup flow and steam production due to boiling in the core. The parameters which act to depressurize the RCS are the flow through the PORV and the condensation of steam due to the cold makeup. The RCS volume balance can be summarized mathematically as

$$Q = Q_m + Q_c - Q_p \qquad \text{(C-1)}$$

where

Q = net volumetric flow,

Q_m = volumetric flow of makeup into the RCS,

Q_c = volumetric production of steam in the core, and

Q_p = volumetric flow out the PORV.

The sign of Q determines if the RCS will depressurize or pressurize, and the magnitude of Q is roughly proportional to the rate of pressure change. The individual terms in the volume balance are computed as

$$Q_m = W_m v_f \qquad \text{(C-2)}$$

$$Q_c = [P - W_m(h_f - h_m)]\, v_{fg}/h_{fg} \qquad \text{(C-3)}$$

$$Q_p = W_p v_p \qquad \text{(C-4)}$$

where

W_m = makeup mass flow rate,

v_f = specific volume of saturated liquid,

P = core power,

h_f = specific enthalpy of saturated liquid,

h_m = specific enthalpy of makeup,

v_{fg} = difference in specific volume between saturated liquid and gas,

h_{fg} = difference in specific enthalpy between saturated liquid and gas,

W_p = PORV mass flow rate, and

v_p = specific volume of the fluid at the PORV.

The above equations are valid for quasi-steady flow in which all the makeup flows past the core. Some of the core power goes into heating the cold makeup to the saturation temperature. The remainder of the power boils liquid to steam. Note that the subcooling of the makeup is assumed to reduce steam production in the core rather than condense steam. However, the net steam production is identical if the makeup is assumed to condense steam rather than reduce steam production in the core. Also note that the expansion of makeup due to heating is accounted for in Equation (C-2) by multiplying by the specific volume of saturated liquid rather than of the cold makeup. The volumetric flow out the PORV is based on the PORV flow area and the fluid state at the PORV. For the volume balance, the PORV area was increased by the effective area of the HPVVs. The PORV flow was calculated with the Henry-Fauske and homogeneous equilibrium critical flow models[C-7] for subcooled and two-phase flow, respectively. The fluid state at the PORV was handled parametrically rather than calculated directly.

A validation of the simple volume balance was performed by comparison with the results of the R586 feed and bleed calculation. The RCS pressure, core power, and makeup flow from the R586 calculation at the time that the RCS depressurization began were used to evaluate Equations (C-2) and (C-3). Equation (C-4) was evaluated as a function of void fraction at the PORV. The volume balance, Equation (C-1), predicted that the RCS should depressurize when the void fraction at the PORV exceeded 0.82. The RCS actually began to depressurize in the R586 feed and bleed calculation when the void fraction reached 0.78. The agreement between the volume balance and the code calculation was considered excellent given the simple assumptions of the volume balance. The small difference between the volume balance and the code calculation was probably caused by the use of different critical flow models, which provided similar but not identical results. The volume balance was also able to predict the key results of the R585 and TRAC calculations. In particular, the volume balance correctly predicted that the R585 calculation described in Section 3.4.1 would continuously depressurize following the initiation of feed and bleed, while the R586 calculation, which had 25% higher power, would not. The volume balance also correctly predicted that the TRAC calculation illustrated in Figure 36 of the main body of this report would not continuously depressurize after the initiation of feed and bleed because of the smaller PORV flow in this calculation (see Section 3.4.2).

The volume balance was used to determine the sensitivity of the calculated depressurization to variations in core power. Figure C-3, which summarizes the results of the volume balance, shows the volumetric flow through the PORV, Q_p (Equation (C-4)), versus quality. Quality, x, was evaluated as

$$x = (h - h_f)/h_{fg} \tag{C-5}$$

where h is the specific enthalpy of the fluid at the PORV. Equation (C-5) yields negative values of qualities for subcooled thermodynamic fluid states. Figure C-3 also shows the combined compressive term, which is the sum of the makeup, Q_m, and core boiling, Q_c, terms [Equations (C-2) and (C-3)]. The core power was varied parametrically as a function of time after reactor trip from 100% rated power. The results shown on the figure were obtained at a pressure of 17.2 MPa (2500 psia), which is close to the pressurizer SRV setpoint pressure, and assumed both makeup pumps were available.

When the volumetric flow through the PORV exceeds the flow due to the combined effects of makeup and core boiling, the PORV can depressurize the RCS. When the PORV flow is less than the compressive term, the RCS will pressurize until the SRVs open. The volumetric flow through the PORV is a strong function of the quality at the PORV. The minimum volumetric flow occurs at a quality near zero. The maximum volumetric flow occurs when the quality is unity and dry, saturated steam flows through the PORV. Superheated fluid states are not shown in the figure, since they imply core uncovering in a quasi-steady analysis.

Figure C-3 shows that the PORV cannot depressurize the RCS for times less than 30 min after reactor trip, regardless of the fluid state at the PORV. The compressive term decreases with time after reactor trip as the decay power and the steam production in the core decrease. At about 37 min after reactor trip, the RCS will depressurize if dry, saturated steam flows through the PORV. At 60 min, about the time depressurization occurred in the R586 feed and bleed calculation, the RCS can depressurize if the quality exceeds 0.65. The RCS will depressurize with almost any fluid at the PORV at 120 min after reactor trip.

The sensitivity of the calculated depressurization to uncertainty in the feed and bleed boundary conditions is illustrated in Figure C-4. The figure shows the best-estimate PORV flow and makeup plus core boiling terms at 60 min after reactor trip from Figure C-3. This time was selected because it was approximately the time that depressurization occurred in the R586 feed and bleed calculation. The figure also shows the best-estimate PORV flow reduced by 20% and the makeup and core boiling terms resulting from a 5% increase

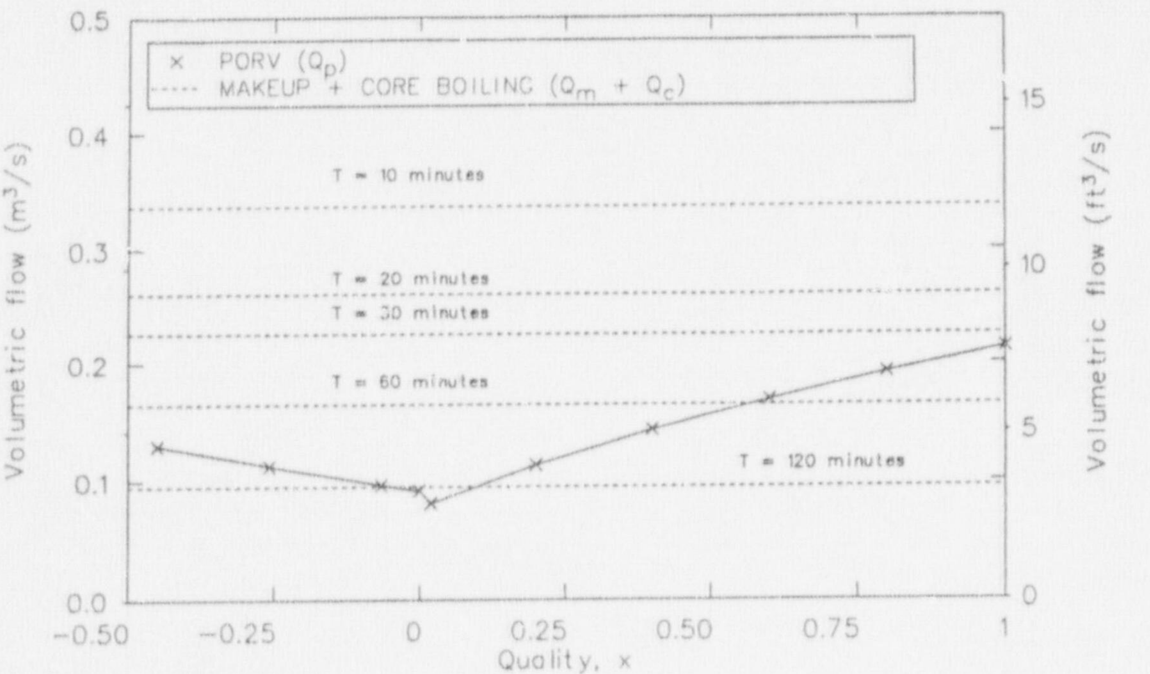

Figure C-3. Results of the volume balance for different times after reactor trip.

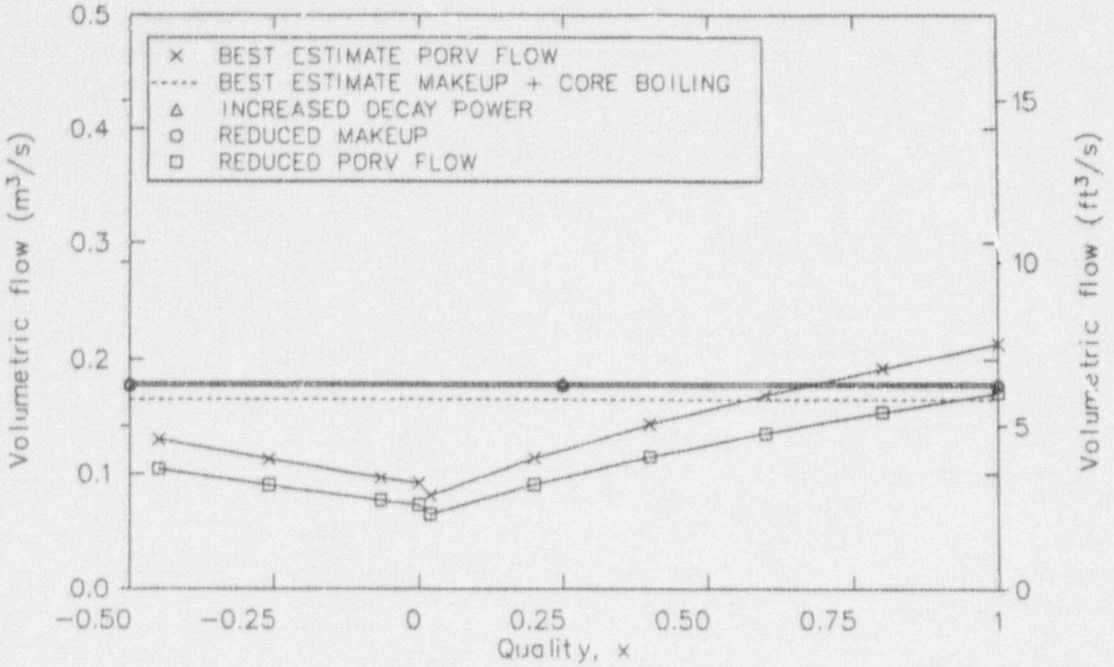

Figure C-4. The effect of uncertainty in boundary conditions on the results of the volume balance.

in core decay power and a 10% decrease in makeup flow. These variations, which correspond to the estimated uncertainties shown in Table C-1, were selected to make depressurization more difficult.

The effects of the uncertainty in decay power and makeup flow were relatively small. The volume balance indicated that the PORV could still depressurize the RCS at 60 min with either the higher core power or the lower makeup flow. However, a slightly higher quality at the PORV would be required before the depressurization could begin. It was estimated that if the core power had been increased by 5% in the R586 feed and bleed calculation, the depressurization would have been delayed by about 100 s. The uncertainty in the volume balance due to the PORV flow was larger than due to the uncertainty in the other two parameters. However, the PORV could still depressurize the RCS once the quality at the PORV approached unity. Thus, the uncertainties in the boundary conditions of the R586 feed and bleed calculation do not have a large effect on the calculated ability to depressurize. The calculated depressurization appears valid and would not be expected to vary significantly because of uncertainty in the boundary conditions. The estimated uncertainty in the time of depressurization due to the combined uncertainty in the three boundary conditions is 5 min.

Although not shown in the figures, the RCS generally depressurizes more easily at lower pressures. This is primarily because as the RCS pressure decreases, the makeup flow increases, which reduces steam production in the core. Thus, the figures shown, which were developed for the maximum possible pressure, corresponding to the SRV setpoint pressure, represent a worst case for depressurization. It should also be pointed out that makeup flow is crucial for the success of feed and bleed in Davis-Besse. Without any makeup flow, the volume balance indicates that PORV would not be able to depressurize the RCS until approximately 140 min after reactor trip. By this time in the R586 feed and bleed calculation, the mixture level probably would have dropped below the top of the core and a core heatup would have begun.

Although the volume balance is useful for understanding the results of code calculations and determining the uncertainty in the calculated depressurization, the method has limitations. In particular, the depressurization time required until RCS refill begins and the minimum liquid level in the vessel are not obtained. Thus, even though the volume balance predicts that depressurization will occur, there is no guarantee that the core will remain covered as the RCS depressurizes. The prediction of depressurization by the volume balance is thus a necessary, but not a sufficient condition, to guarantee the success of feed and bleed.

C-2.4 Code User. The code user is also a potential contributor to the uncertainty in a calculation. However, the user affects the calculation primarily through the selection of the thermal-hydraulic computer code, the code input model, initial conditions, and boundary conditions. Since the uncertainty related to these potential contributors was described previously, no significant, additional uncertainty was thought to be related to the code user.

C-2.5 Assumed Transient. The assumed transient can have a large effect on the calculated results. The initiating event, equipment performance, and operator actions strongly influence the course of a transient. Thus, significantly different results could be calculated if different assumptions were made relative to these parameters. Examples of parameters which significantly affect a feed and bleed transient are the feedwater flow and the makeup flow. If some feedwater flow continues after reactor trip, such as occurred in the Davis-Besse LOFW transient of June 9, 1985, the OTSG dryout can occur much later than was obtained in the R586 feed and bleed calculation. A later OTSG dryout delays the time when the RCS reaches a certain temperature and possibly the initiation of feed and bleed. Conversely, if less feedwater is delivered to the OTSGs than in the R586 calculation, higher RCS temperatures result at the initiation of feed and bleed. For example, in the R586 feed and bleed calculation, the initiating event, the main feed pump (MFP) trip, was assumed to result in immediate turbine and reactor trips. If reactor trip on MFP trip failed, the reactor trip would be delayed until high RCS pressure or temperature conditions occurred. The reactor trip would then occur at a much lower OTSG liquid inventory than in the R586 calculation. In this new transient scenario, OTSG dryout would occur earlier, the RCS would be at a higher temperature, and would possibly even be two-phase when feed and bleed was initiated, and RCS voiding and depressurization would occur more rapidly than was obtained in the R586 feed and bleed calculation.

The assumed behavior of the makeup system can also have a large effect on the calculated temperature of the RCS at the initiation of feed and bleed. In the R586 calculation, no makeup flow was assumed prior to the initiation of feed and bleed. If a more realistic scenario had been modeled, the operators would have manually started the second makeup pump, providing maximum flow until the pressurizer level was recovered. In this scenario, the RCS temperatures at 20 min would have been about 5 K (9°F) lower than in the R586 calculation. Note that the variation due to the assumed makeup response was as large as the total estimated uncertainty in RCS temperature due to initial and boundary conditions. If maximum makeup for the entire transient was assumed, the RCS temperature would have been about 17 K (30°F) lower at the initiation of feed and bleed than in the R586 calculation. Thus, the sensitivity of the calculated results to assumptions regarding the initiating event and operator actions can be much larger than the uncertainty in the calculations due to the sources described in this report. The uncertainties in this report only apply to the calculation as performed, given the assumed transient, equipment performance, and operator actions.

C-3. Uncertainty in Important Parameters

The important parameters for a feed and bleed transient were identified as the RCS temperature at the time feed and bleed was initiated, the depressurization during feed and bleed, and the minimum liquid level in the reactor vessel. The estimated uncertainty in the calculation of the important parameters is discussed below.

The uncertainty in the RCS temperature when feed and bleed was initiated in the R586 calculation is estimated to be 5 K (9°F). This uncertainty was caused by uncertainty in the initial and boundary conditions. The largest contributor to this uncertainty was the initial liquid inventory in the OTSGs. The stated uncertainty bounds the deviations between the calculated and measured temperatures for the Davis-Besse LOFW transient of June 9, 1985 (see Figures 11 and 12 from the main body of this report). The stated uncertainty is also representative of the deviation observed between the assessment calculation and data from a feed and bleed test in an experimental facility scaled to a B&W PWR (see Reference C-2).

The uncertainties in the R586 calculation due to initial and boundary conditions are also applicable to the R585 calculations. However, the uncertainties in the R585 calculations are larger because of the erroneous 25% reduction in core power when the RCPs tripped, as discussed in Section 3.4.1. The reduction in core power resulted in a calculated heatup rate that was about 25% too low after RCP trip. This error caused a bias in the R585 calculations in addition to the uncertainty associated with the initial and boundary conditions. For the R585 calculations described in this report, the error caused the calculated RCS temperature to be about 2 K (4°F) too low at the initiation of feed and bleed. The time required to reach the RCS temperature at which feed and bleed was initiated was about 1 min too long. The bias was larger in those R585 calculations, not described in this report, in which feed and bleed was initiated at 37 min. For these calculations, the errors in RCS temperature at the start of feed and bleed and the time to reach this temperature were about 11 K (20°F) and 6 min, respectively.

The uncertainty in the boundary conditions of core decay power, PORV flow, and makeup flow does not significantly alter the results of the R586 calculation relative to the depressurization of the RCS. The uncertainty in the time at which the RCS depressurization began due to uncertainty in the boundary conditions was estimated to be 5 min. The calculated time of depressurization did not appear to be sensitive to hot leg nodalization. The uncertainty in the depressurization in the R585 calculations was thought to be much larger for those feed and bleed calculations in which the RCPs tripped.

The uncertainty in the calculated collapsed liquid level in the reactor vessel was estimated to be 1 m (3 ft). This result was based on the results of assessment calculations. In the R586 feed and bleed calculation, an uncertainty was identified relative to the RCP nodalization which could cause the calculated level to be about 0.3 m (1 ft) too high. The uncertainty in the level could also be larger because of the upper head modeling for those transients in which feed and bleed was initiated at relatively low RCS temperatures. The uncertainty in the R585 calculation may be slightly larger because the small holes in the plenum cylinder at the hot leg nozzle elevation were not explicitly modeled. However, the uncertainty due to the upper head and upper plenum models should not affect the calculated result that the minimum collapsed liquid level remains above the top of the core.

The above estimates of uncertainty in the important feed and bleed parameters are valid for the transients analyzed based on the assumed initiating event, equipment performance, and operator actions. Significantly different results could be calculated if different assumptions were made regarding these parameters. In particular, different assumptions regarding feedwater flow, makeup flow, and reactor trip could significantly alter the calculated results.

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NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-4946
EGG-2510

2. TITLE AND SUBTITLE
Davis-Besse Uncertainty Study

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
August	1987

5. AUTHOR(S)
Cliff B. Davis

6. DATE REPORT ISSUED

MONTH	YEAR
August	1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
FIN No. A6827

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

The uncertainties of calculations of loss-of-feedwater transients at Davis-Besse Unit 1 were determined to address concerns of the U.S. Nuclear Regulatory Commission relative to the effectiveness of feed and bleed cooling. Davis-Besse Unit 1 is a pressurized water reactor of the raised-loop Babcock & Wilcox design. A detailed, quality-assured RELAP5/MOD2 model of Davis-Besse was developed at the Idaho National Engineering Laboratory. The model was used to perform an analysis of the loss-of-feedwater transient that occurred at Davis-Besse on June 9, 1985. A loss-of-feedwater transient followed by feed and bleed cooling was also calculated. The evaluation of uncertainty was based on the comparisons of calculations and data, comparisons of different calculations of the same transient, sensitivity calculations, and the propagation of the estimated uncertainty in initial and boundary conditions to the final calculated results.

FIN No. A6827—International Code Assessment: RELAP5 and TRAC-BWR

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

120555078877 1 1AN1R4
US NRC-OARM-ADM
DIV OF PUB SVCS
POLICY & PUB MGT BR-PDR NUREG
W-537
WASHINGTON DC 20555

EG&G Idaho
P.O. Box 1625
Idaho Falls, Idaho
83415