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INFORMAL REPORT

N REACTOR RELAP5 MODEL BENCHMARK COMPARISONS

C. D. Fletcher
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Idaho National Engineering Laboratory

Managed by the U.S. Department of Energy

EG&G Idaho

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ABSTRACT

This report documents work performed at the Idaho National Engineering Laboratory (INEL) in support of Westinghouse Hanford Company safety analyses for the N Reactor. The portion of the work reported here includes comparisons of RELAP5/MOD2-calculated data with measured plant data for: (1) a plant trip reactor transient from full power operation, and (2) a hot dump test performed prior to plant startup. These qualitative comparisons are valuable because they provide an indication of the capabilities of the RELAP5/MOD2 code to simulate operational and blowdown transients in the N Reactor.

SUMMARY

This report documents analyses performed at the Idaho National Engineering Laboratory (INEL) in support of Westinghouse Hanford Company (WHC) safety analysis for the N Reactor. These analyses support a spectrum of safety analyses performed at the INEL for WHC. The work reported here consisted of performing RELAP5/MOD2 computer code calculations of actual N Reactor transients and comparing the results of the code-calculated results with the data from the plant. The purpose of these calculations was to provide a qualitative assessment of the capabilities of the computer code and plant model for simulating thermal-hydraulic behavior during transient sequences.

The accident sequences to be investigated as a part of the planned safety analysis are generally much more challenging simulation problems than the transients for which plant data are available. Nevertheless, the application of the code and model to these less-challenging transients provides a valuable limited assessment of simulation capabilities. These benchmark calculations represent a logical first step toward gaining confidence in the computer-generated results for the safety analysis sequences.

Benchmark comparisons between code-calculated and measured plant data were performed for two plant transients: (1) a reactor trip sequence starting from full power operation, and (2) a hot dump test performed prior to startup of the plant.

In the reactor trip sequence, all plant systems performed as designed and the primary coolant system remained liquid-filled. The sequence included a minor plant cooldown and exercised many of the plant control systems. Calculated and measured data for pressurizer pressure compared well. The comparison for pressurizer level was adequate. These comparisons indicate there likely is more lag in the plant pressurizer level indication than has been previously thought. The HPI flow rate comparison was poor, but there is uncertainty concerning the plant data for this parameter.

In the hot dump test sequence, the primary coolant system was heated to elevated pressure and temperature by the action of the primary coolant pumps; no core power was used. The test was initiated by tripping the primary coolant pumps. ECCS actuation, due to low core flow and low primary coolant pump speed, caused the blowdown (V-4) valves to open. The primary coolant system depressurized as fluid exited through the V-4 valves and dump line into the dump basin. Calculated and measured data for system pressures and dump line mass flow rate compared very well. The comparison for primary coolant pump speed was acceptable. A sensitivity of the calculated results to the modeled pump bearing friction was identified. Minor differences between calculated and measured event sequence timing were attributed to the effect of bearing friction on the pump coastdown characteristics.

The comparisons of calculated and measured data indicated the RELAP5 computer code and model adequately simulate the thermal-hydraulic phenomena for operational transients involving a single-phase primary coolant system and for blowdown transients involving a two-phase primary coolant system. Furthermore, these comparisons have provided a satisfactory global check of modeled system volumes, heat transfer areas, metal masses, flow characteristics, and control system processes.

ACKNOWLEDGMENTS

The authors wish to thank the following people for their contributions to this report: S. R. Wagoner, S. A. Wood, and J. M. McLaren of Westinghouse Hanford Company for their valued input to the understanding of the operation of the N Reactor and to the construction of the RELAP5 input model of the N Reactor. The authors thank D. M. Ogden, also of Westinghouse Hanford Company, for his timely reviews and comments of the analyses performed at EG&G Idaho for the N Reactor.

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N REACTOR RELAP5 MODEL BENCHMARK COMPARISONS

1. INTRODUCTION

The N Reactor is operated by the Westinghouse Hanford Company at Richland, Washington for the U. S. Department of Energy. The reactor is of graphite-moderated, light-water-cooled, pressure-tube design with a rated capacity of 4000 MW_t. This report documents work performed at the Idaho National Engineering Laboratory (INEL) in support of Westinghouse Hanford Company (WHC) safety analyses for the N Reactor. This document reports two benchmark calculations, which support a spectrum of nine safety analyses performed at the INEL for WHC. The other analyses include cold leg manifold breaks[1,2], station blackout sequences[1,2], and inlet and outlet riser breaks[3].

The N Reactor core is a cuboid, 33 feet by 33 feet at the face and 39 feet long. Interlocking graphite bars support pressure tubes which house the fuel elements. A total of 1003 horizontal pressure tubes, running from front to rear, penetrate the graphite moderator. The reactor coolant flows through the pressure tubes and transfers the heat from the low-enrichment, metalic-uranium, tube-in-tube fuel elements to the secondary coolant in the steam generators.

The reactor coolant system consists of 16 parallel piping runs that conduct cooling water from a cold leg manifold to the reactor core. Each of these 16 pipes terminates in a vertical riser. Connected to each riser are 54 to 66 individual pressure tubes. Similar hot leg risers and parallel lines transport the coolant from the pressure tubes to a hot leg manifold. The hot and cold leg manifolds are connected to six parallel steam generator cells. Each cell consists of two parallel steam generators, a primary coolant pump and associated piping, valves and instrumentation. During normal operation, only five of the cells are used.

In the primary coolant loops there are three groups of check valves to assure that coolant does not bypass the core during an accident. The location of these valves are at the primary pump discharge (PCSV-202), upstream of the core (CV-5), and upstream of the hot leg manifold (CV-3).

Other primary coolant system components include the emergency core cooling system (ECCS), and the high pressure injection system (HPI). The ECCS provides a separate independent water system for emergency once-through cooling of the reactor core. The HPI system provides normal makeup for the system during operation. The HPI system also provides fluid to the system for pressure and pressurizer level control.

Figure 1 shows a pictorial view of the primary coolant system of the N Reactor.

The work reported here consisted of performing RELAP5/MOD2 computer code calculations of two plant transients: (1) a manual plant trip from full power operation, and (2) a hot dump startup plant test. The computer calculations of these transients are compared against the actual measured plant data for the purpose of evaluating the fidelity of the computer simulations. Section 2 presents a description of the computer code used in the analyses. Section 3 presents a description of the N Reactor computer models used for these calculations. Sections 4 and 5 present the results of the plant trip and hot dump test analyses respectively. Section 6 presents analysis conclusions; references are given in Section 7.

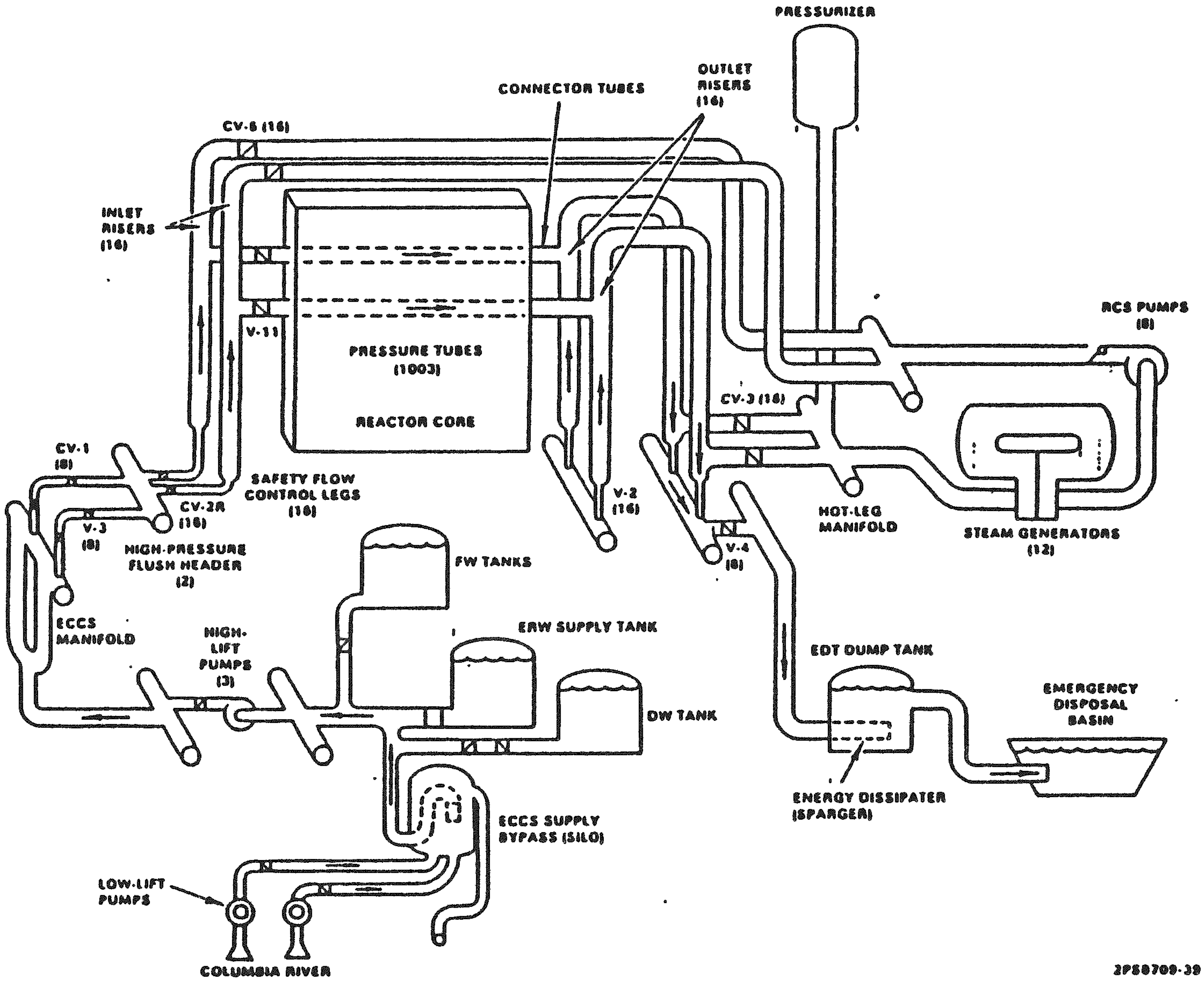

Figure 1. Layout of the N Reactor.

2. CODE DESCRIPTION

This section presents a description of the computer code used in this analysis. RELAP5/MOD2[4] is a light water reactor transient analysis code that has been developed at the Idaho National Engineering Laboratory (INEL) for the U. S. Nuclear Regulatory Commission (USNRC). The core is currently being applied for safety analyses of the U. S. Department of Energy nuclear reactors. The mission of the code is to provide an advanced best-estimate predictive capability for applications pertaining to nuclear power plant safety. Examples of this capability have included analytical support for: the LOFT and Semiscale experimental programs, the relief valve testing program, simulation of design basis loss of coolant accidents, the pressurized thermal shock investigation, and evaluation of operational transients and operator guidelines in light water reactor systems. RELAP5 is a generic code that can be used for simulation of a wide variety of thermal-hydraulic transients in both nuclear and nonnuclear systems involving steam-water-noncondensible fluid mixtures.

The RELAP5 code solves the mass, momentum, and energy conservation equations for both the liquid and vapor phases. It is a fully nonhomogeneous and nonequilibrium code. The equations are solved by a fast, partially-implicit numerical scheme to permit economical calculation of system transients. The objective of the development effort has been to produce a code that includes important first-order effects necessary for accurate prediction of system transients, but yet is sufficiently simple and cost effective so that parametric or sensitivity studies are possible.

RELAP5 utilizes generic component models from which general thermal-hydraulic systems can be simulated. The component models include pumps, valves, pipes, heat structures, point reactor kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control system elements. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and a noncondensible gas.

The development of RELAP5 has spanned approximately twelve years from the early stages of numerical scheme development to the present. The code represents the aggregate accumulation of experience in modeling two-phase processes, and light water reactor systems in particular. The code development has benefited from extensive application and comparison to experimental data. Additional experience has been gained through use of the code by many research and development institutions in the U.S. and several foreign countries.

The computer code used for the analyses presented in this report was RELAP5/MOD2, Version 36.05. This version was the production code at the time the calculations presented here were performed.

In analysis of N Reactor accident sequences involving reflooding of the horizontal reactor core, the production version of the RELAP5/MOD2 computer code did not adequately simulate the core refill phenomena. Code modifications were made to support analysis of the reflood portions of such sequences, and a description of these modifications is presented in Section 2.1 of Reference 1. Since the plant transients simulated in this report did not involve reflooding of the core, the production version of RELAP5/MOD2 was used to calculate these transients in their entirety.

3. MODEL DESCRIPTION

The RELAP5/MOD2 model[5] of the N Reactor was developed by converting an existing RETRAN model[6] of the plant. The RELAP5 model represents the reactor core, primary coolant loop and associated systems (high pressure injection, emergency core cooling, etc.), and the steam generators. A detailed discussion of the basic RELAP5 N Reactor model is given in Section 3 of Reference 1.

Section 3.1 describes the RELAP5 N Reactor model used for the plant trip transient calculation and Section 3.2 describes the model used for the hot dump test transient calculation. Listings of the RELAP5 models used for the plant trip and hot dump test transient calculations are presented in Appendices A and B respectively.

3.1 Model for the Plant Trip Calculation

The model used for the plant trip calculation was the same as the lumped-core RELAP5 model described in Section 3 of Reference 1. The model included a single-channel model of the reactor core region and a single loop model representing the five active steam generator cells. A nodalization diagram of the model used for the plant trip calculation is shown in Figure 2. The simplified reactor core nodalization was adequate for calculating the plant trip transient because no voiding occurred within the reactor core during the event sequence.

Steady state conditions obtained with the RELAP5 model are compared with measured plant data at the time of the plant trip in Table 1. Good agreement is indicated between the limited available plant measured data and the code calculated initial conditions. The initial plant state for the plant trip transient was steady 3870 MW_t operation. A description of the plant trip transient appears in Section 4.1.

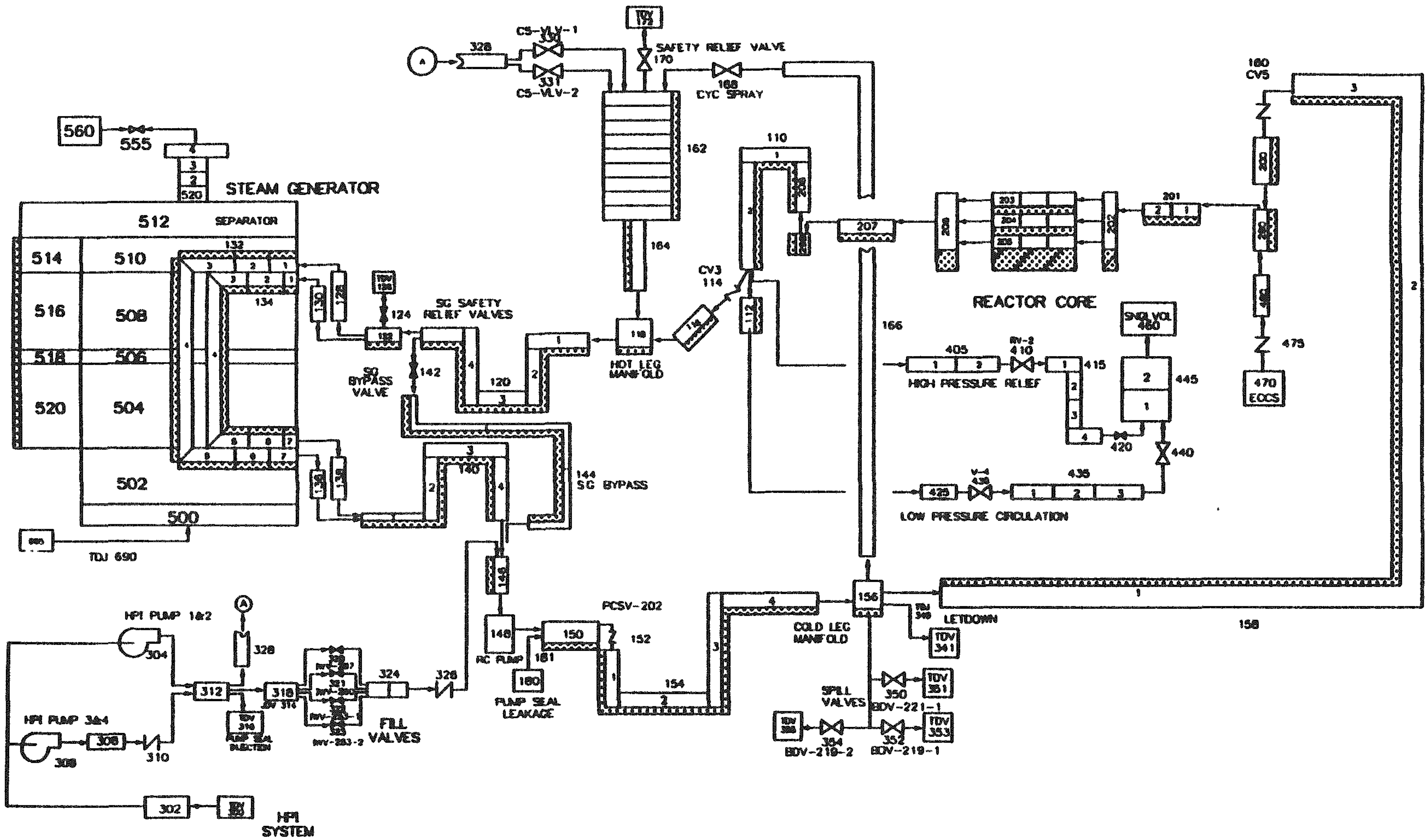

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Figure 2. Nodalization of the RELAP5 N Reactor model for the plant trip calculation.

TABLE 1. COMPARISON OF RELAP5-CALCULATED STEADY STATE WITH PLANT DATA FOR THE PLANT TRIP ANALYSIS

Parameter (units)	Plant Data	RELAP5
Core Power (MW_t)	3870	3870
Pressurizer Pressure (psia)	1595	1595
Pressurizer Level (ft)	24.5	24.4
Hot Leg Temperature (^oF)	NA	532
Cold Leg Temperature (^oF)	NA	394
Loop Flow Rate (lbm/s)	NA	23745
Steam Line Pressure (psia)	152	152
Feedwater Flow Rate (lbm/s)	NA	3729
HPI Flow Rate (lbm/s)	800	979
Core Differential Pressure (psid)	NA	111
S.G. Primary Differential Pressure (psid)	NA	27

NA = Not Available

3.2 Model for the Hot Dump Test Calculation

The model used for the hot dump test calculation was the same as the multiple-channel core model described in Section 3 of Reference 1. The model included an eight-channel representation of the 1003 core process tubes and a lumped representation of the active steam generator cells. The 8-channel core nodalization, shown in Figure 3, was inserted into the coolant loop nodalization of Figure 2 to form the model used for this calculation.

The hot dump plant startup test was performed with only four active steam generator cells; normal operation uses five active cells. Therefore, the loop model (see Section 3 of Reference 1), from the hot leg manifold to the cold leg manifold, was modified to represent the four active steam generator cells.

Measured plant conditions at the beginning of the hot dump test are compared with RELAP5-calculated steady initial conditions in Table 2. For the limited plant parameters available, RELAP5-calculated and measured plant data are in good agreement. The test was performed with no core power; the primary coolant system was uniformly heated by operation of the primary coolant pumps. A description of the hot dump test appears in Section 5.1.

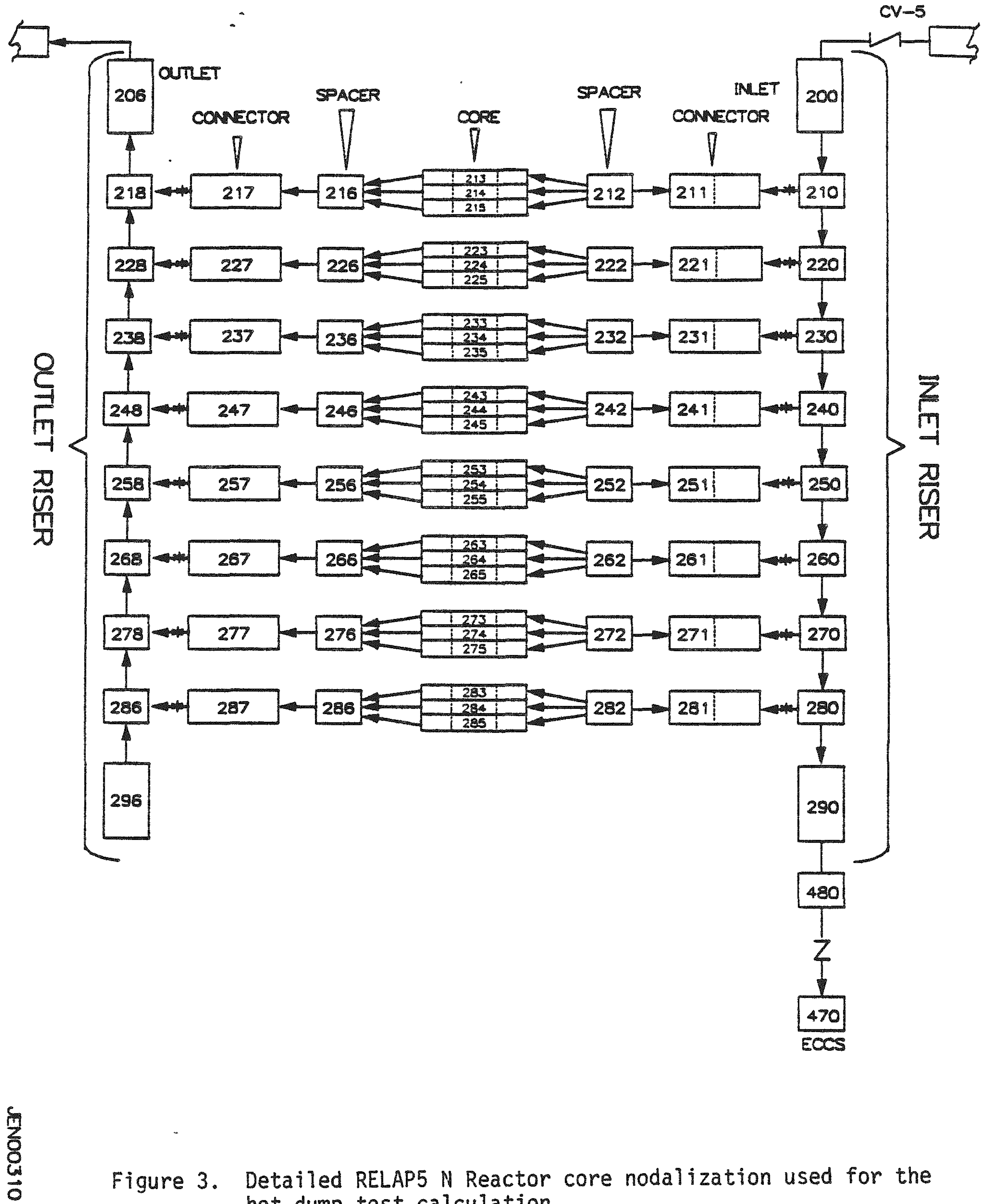

Figure 3. Detailed RELAP5 N Reactor core nodalization used for the hot dump test calculation.

TABLE 2. COMPARISON OF RELAP5-CALCULATED STEADY STATE WITH PLANT DATA FOR THE HOT DUMP TEST ANALYSIS

Parameter (units)	Plant Data	RELAP5
Core Power	0	0
Cold Leg Riser Pressure (psia)	1436	1440
Pressurizer Level (ft)	24.8	25.1
Hot Leg Temperature (^oF)	450	451
Primary Coolant Pump Speed (rpm)	3200	3200
Loop Flow Rate (lbm/s)	18600	17477
Steam Line Pressure (psia)	NA	14.7
Steam Generator Liquid Mass	0	0
Feedwater Flow Rate	0	0
HPI Flow Rate	0	0
Core Differential Pressure (psid)	NA	77
S.G. Primary Differential Pressure (psid)	NA	20
ECCS Liquid Temperature (^oF)	122	122

NA = Not Available

4. PLANT TRIP ANALYSIS

A plant trip occurred in the N Reactor on January 12, 1980. A comparison of code-calculated and measured plant data for this transient is valuable as a qualitative indicator of the RELAP5 computer code and model capabilities to simulate a simple plant transient. The comparison provides an overall check on modeled system volumes, heat transfer areas, metal masses, and control processes. Measured plant data during the plant trip are provided in Reference 7. Section 4.1 describes the plant trip event and Section 4.2 presents the results of the code-data comparison analysis.

4.1 Transient Description

The N Reactor was operating at 3870 MW_t on January 12, 1980 when a fuel rupture indication occurred in a process tube. In accordance with operating procedures, the reactor was manually tripped in response to the fuel rupture indication.

The operating conditions at the time of the reactor trip were previously presented in Table 1. These conditions represent normal full power operation.

All plant systems operated as designed following the reactor trip. Steam generator primary system bypass valves were opened. Feedwater was controlled to maintain the setpoint steam generator level.

In response to the reactor trip, the primary coolant system pressure and average fluid temperature decreased. The coolant shrinkage caused the pressurizer level to decrease and this resulted in increased high pressure injection (HPI) flow. HPI flow was modulated when the pressurizer setpoint level was lowered at 59 s. At 110 s the pressurizer setpoint pressure was also lowered and this caused initiation of cold pressurizer spray and a pressure decrease. The changes in pressurizer level and pressure setpoints are designed to minimize pressure fluctuations following a reactor trip and shorten the time required to stabilize the primary coolant system.

4.2 Code-Data Comparison Analysis

A RELAP5/MOD2 transient calculation was performed for the first 180 s of the plant trip sequence. The calculation was initiated from the steady conditions presented in Table 1. As previously discussed, good agreement was obtained between code-calculated and limited available plant measured initial condition data.

The RELAP5-calculated sequence of events during the plant trip simulation is compared with the measured events in Table 3.

Calculated and measured pressurizer level responses for the plant trip are compared in Figure 4. Pressurizer level initially decreased because, following the reactor trip, the average primary coolant system temperature declined, shrinking its volume. As the pressurizer level decreased, the steam bubble at the top of the pressurizer expanded, causing the primary coolant system pressure to decrease as well. Figure 5 compares the calculated and measured pressurizer pressure responses. Also in response to the decreasing pressurizer level, the high pressure injection system flow increased. Calculated and measured high pressure injection flow rate responses are shown in Figure 6.

The fluid volume added by the increased HPI flow counteracted the volume shrinkage caused by the declining fluid temperatures. At about 60 s in both the calculated and measured data the HPI volume addition rate equaled the fluid shrinkage rate, creating minimums in pressurizer level and pressure responses.

The changes in both calculated and measured pressure responses at about 110 s were caused by the scheduled reduction in pressurizer setpoint pressure. This scheduled reduction caused initiation of cold pressurizer spray to lower the pressure to its new, post-trip setpoint.

The measured and calculated pressurizer level responses shown in Figure 4 compared acceptably. The minimum calculated level was about 3 feet

TABLE 3. MEASURED AND CALCULATED SEQUENCE OF EVENTS FOR THE PLANT TRIP

Time (s)		
Measured Plant Data	RELAP5-Calculated Data	Events
0	0	Manual reactor trip.
59	60	Pressurizer level setpoint reduction.
60	60	Minimum pressurizer pressure reached.
78	60	Minimum pressurizer level reached.
64	60	Maximum HPI flow reached.
91-100	110	Pressurizer pressure setpoint reduction.
111	110	Maximum recovery pressure reached.
-	180	RELAP5 calculation terminated.

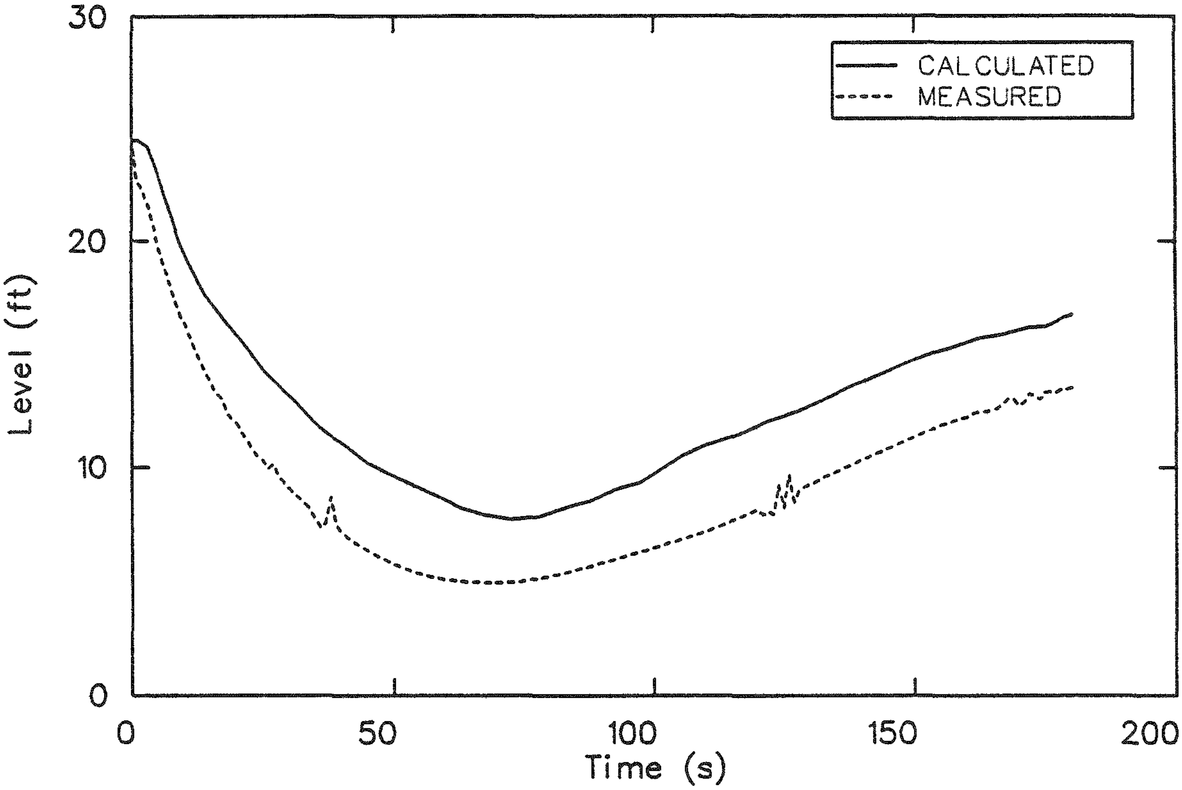

Figure 4. Measured and calculated pressurizer level responses for the plant trip transient.

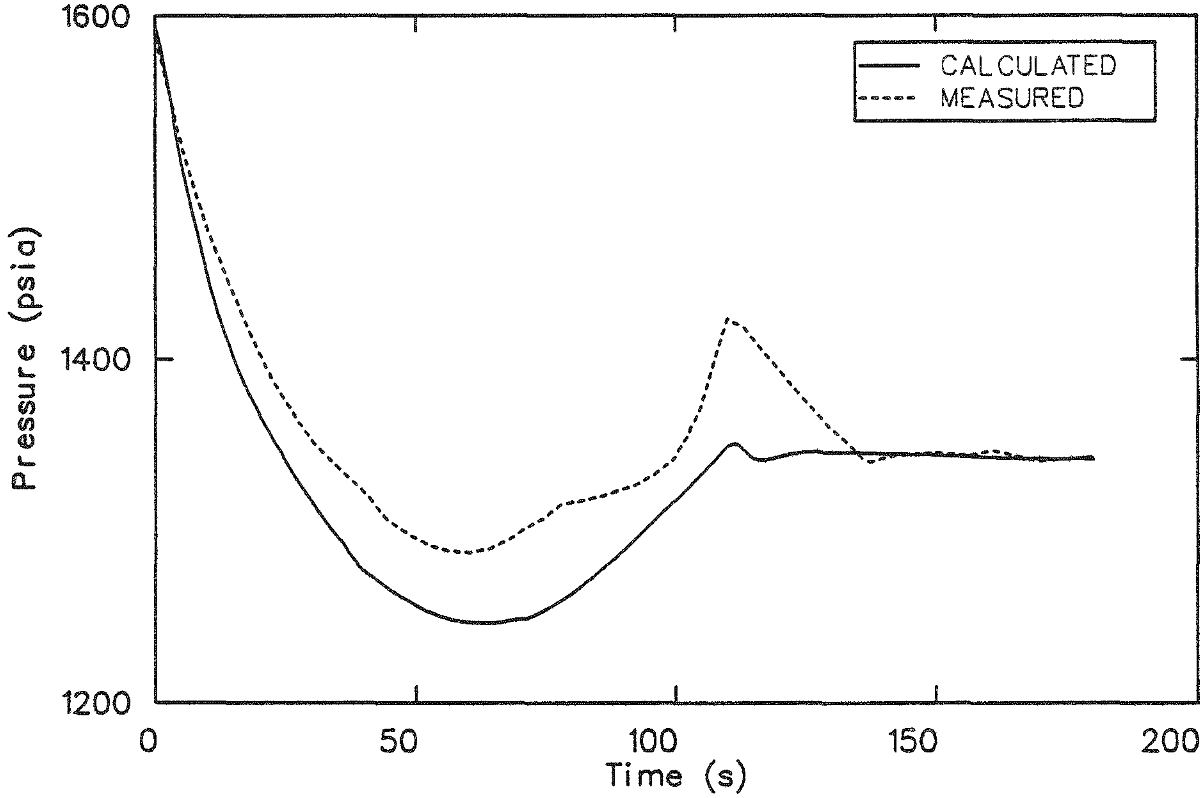

Figure 5. Measured and calculated pressurizer pressure responses for the plant trip transient.

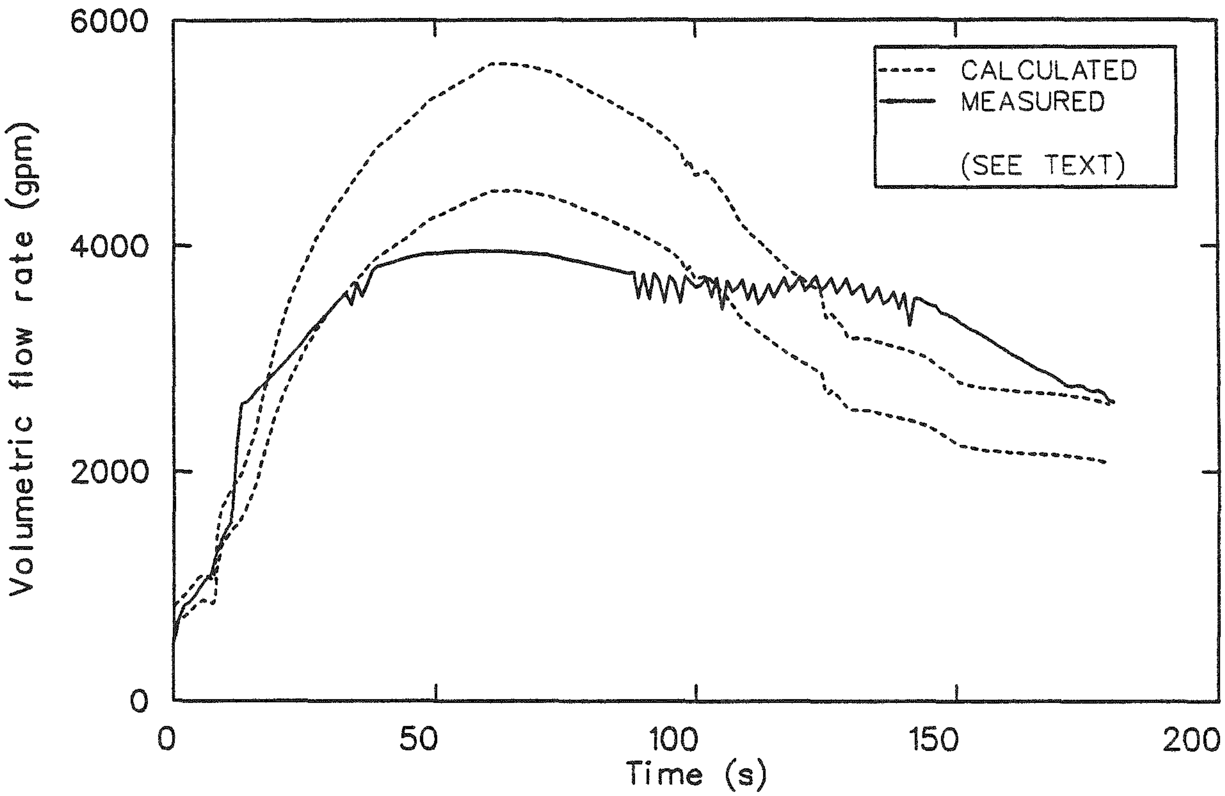

Figure 6. Measured and calculated HPI flow rate responses for the plant trip transient.

below the minimum measured level. The minor fluctuations in the calculated pressurizer level response, at about 40 and 125 s, are believed to be caused by localized flashing and condensation phenomena within the pressurizer. These phenomena momentarily affect the pressurizer indicated level, but not the water level within the pressurizer.

The measured and calculated pressurizer pressure responses shown in Figure 5 compared very well. The trends of the pressure responses and the minimum pressures are in good agreement. Minor short-term differences in the pressure responses are observed in both the depressurization and repressurization phases of the transient.

An anomaly between the measured pressurizer level and pressure responses is also observed. Theoretically, in a liquid-filled system with a steam bubble at the top of the pressurizer, the level and pressure responses should track together. In the case of the calculated level and pressure, the minimums in both were attained simultaneously at 60 s. In the case of the measured level and pressure, however, the minimum level was reached at about 78 s while the minimum pressure occurred at 60 s. The differences in the measured level and pressure responses therefore suggest the pressurizer indicated level may be lagged far more than is currently thought. A longer lag period would explain the timing discrepancies between the calculated and measured pressurizer level responses shown in Figure 4.

The measured and calculated HPI flow rate responses shown in Figure 6 compared well in some periods and poorly in others. The calculated HPI flow generally tracked the measured data (the upper dashed line in Figure 6). However, significant differences were observed in the timing of flow trends and the maximum flow achieved. Previous analyses of this transient[6] have indicated there was significant uncertainty in the plant data for HPI flow rate. In fact, this uncertainty extended to the HPI flow rate during steady operation prior to the reactor trip (see Table 1). The previous analysis concluded the measured HPI flow rate was excessive and that 80% of the measured data approximated the actual flow during the plant transient. The lower dashed line in Figure 6 represents the 80% flow rate. Because of the uncertainty in the plant data, it can not be concluded if the differences between RELAP5-calculated and measured HPI flow rates are meaningful.

In summary, measured and calculated responses for pressurizer pressure compared very well for the January 12, 1980 plant trip event. The calculated and measured pressurizer level responses compared adequately. The level and pressure comparisons suggest the plant indicated level may be lagged more than is now thought. Because of uncertainties in the measured HPI flow rate, the comparison of calculated and measured responses for this parameter are inconclusive. The anomalies observed between calculated and measured responses are considered minor. The results of this code-data comparison are very favorable when compared to similar previous efforts with commercial light-water reactor models. Qualitatively, this indicates the RELAP5/MOD2 code and the N Reactor model adequately simulate the thermal-hydraulic characteristics of the plant for transients involving a single-phase primary coolant system. Furthermore, a satisfactory global check of modeled system volumes, heat transfer areas, metal masses, flow characteristics, and control system processes has been provided by the code-data comparison.

5. HOT DUMP TEST ANALYSIS

During startup testing of the N Reactor in 1964 a hot dump test was performed primarily to evaluate primary coolant system blowdown phenomena, and to assess the ECCS activation criterion and its once-through flow characteristics through the system. The open dump basin has since been replaced with one of covered design.

A comparison of code-calculated and measured plant data for the hot dump test is valuable as a qualitative indicator of the RELAP5 computer code and model capabilities to simulate a blowdown transient involving two-phase conditions in the primary coolant system. Measured plant data during the hot dump test is provided in References 8 and 9. Section 5.1 describes the hot dump test and Section 5.2 presents the results of the code-data comparison analysis.

5.1 Transient Description

The hot dump test was performed on March 27, 1964 as a part of N Reactor startup testing. The test was performed with no core power; the primary coolant system was uniformly heated to 450^{o}F by the action of the primary coolant pumps. Primary coolant pumps were powered at a constant 3200 rpm. Only four of the six steam generator cells were operating at the time of the test; the inactive cells were valved out. Normal operation uses five of the six cells.

Steam generator secondaries were dry, no feedwater was used. The steam generator primary-side bypass valves were closed. Pressurizer heater and spray functions were deactivated. During the blowdown portion of the test, there were no HPI and letdown flows.

The steady state conditions from which the hot dump test began are presented in Table 2. The conditions at the beginning of the test included a flowing primary coolant system at near normal pressures and temperatures.

Normal emergency core coolant system (ECCS) trip logic was used during the test. An ECCS trip resulted when two of the following three conditions were met simultaneously: (1) primary coolant system pressure below 375 psia, (2) primary coolant pump speed below 700 rpm, and (3) core flow below 2 lbm/s per process tube. In the hot dump test sequence, conditions (2) and (3) provided the ECCS trip.

The test was initiated by manually tripping the four operating primary coolant pumps simultaneously. The primary coolant pump coastdown caused flow through the core to slow until the ECCS trip was encountered. ECCS actuation caused the V-4 (blowdown) valves and the V-3 (ECCS injection) valves to open. The primary coolant system depressurized as fluid exited through the V-4 valves and dump line into the dump basin. When the core pressure fell below the ECCS pump shutoff head, ECCS injection to the core began.

The test therefore provided the needed data on the blowdown characteristics of the primary coolant system and ECCS activation verification and its once-through flow characteristics through the system.

5.2 Code-Data Comparison Analysis

A RELAP5/MOD2 transient calculation was performed over the first 350 s of the hot dump test sequence. The calculation was initiated from the steady conditions presented in Table 2. As previously discussed, good agreement was obtained between code-calculated and measured plant data for the test initial conditions.

The RELAP5-calculated event sequence during the hot dump test simulation is compared with the measured event sequence in Table 4.

The calculated and measured pressures at the top of the inlet riser are compared in Figure 7. In this and remaining comparisons, the plant data is represented by the points shown. Plant data collection was performed at infrequent time intervals and much of the data was recorded manually.

TABLE 4. MEASURED AND CALCULATED SEQUENCE OF EVENTS FOR THE HOT DUMP TEST

Time (s)		
Measured Plant Data	RELAP5-Calculated Data	Event(s)
0	0	Test initiated by tripping of primary coolant pumps.
70	48	ECCS signal generated by coincident low primary coolant pump speed and low core flow.
100	80	Pressurizer empty.
***	140	Primary coolant pump coastdown completed.
285	316	ECCS flow begins when core pressure declines below ECCS pump shutoff head.
-	350	RELAP5 calculation terminated.

*** During the test, the four primary coolant pumps responded differently. The pump in Cell 1 stopped turning at 160 s. Pumps in the other cells continued pinwheeling at low speed in the coolant flow.

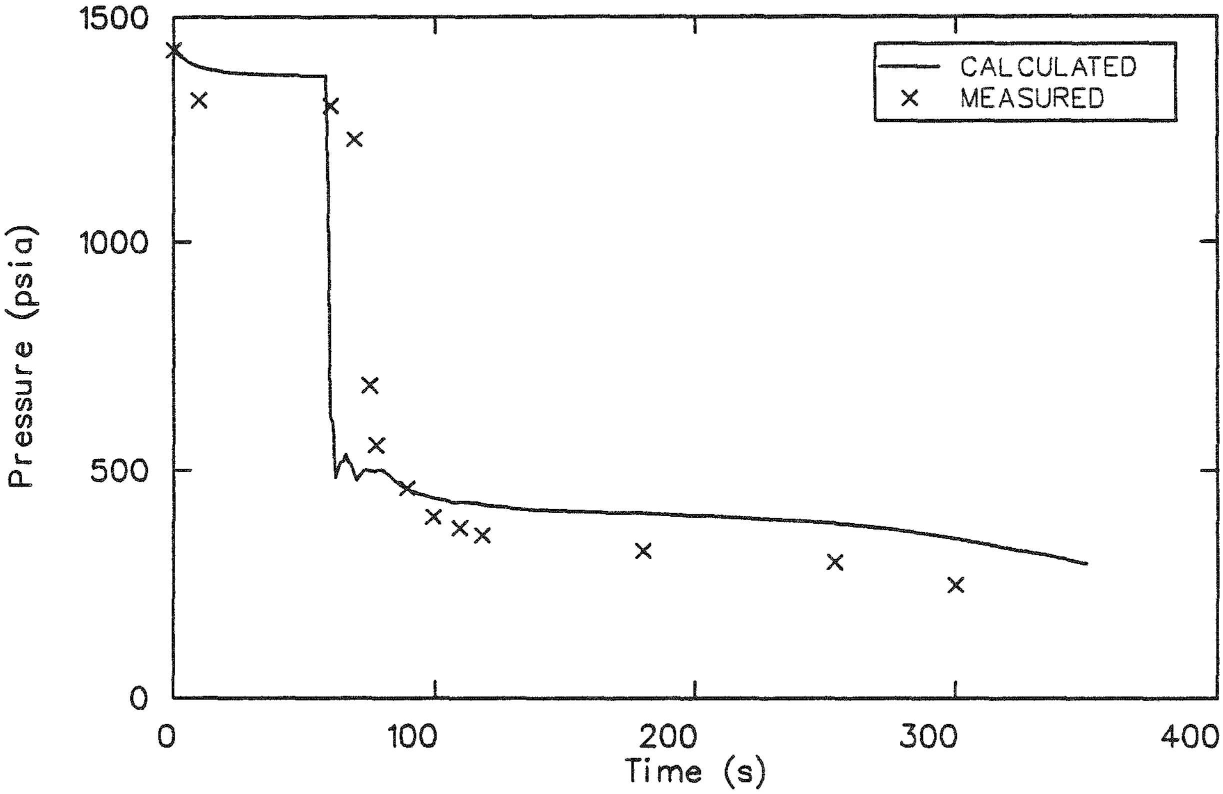

Figure 7. Measured and calculated inlet riser pressure responses for the hot dump test.

The small initial decline in pressure resulted from the loss of primary coolant pump head after the pumps were tripped. The rapid decline in pressure at about 70 s resulted from the opening of the blowdown (V-4) valves.

The calculated and measured pressures downstream of the blowdown valves are compared in Figure 8 and the pressures inside the dump basin sparger are compared in Figure 9. At the beginning of the test, these regions of the dump line were at atmospheric pressure. The increase in pressure at about 70 s was caused by the opening of the blowdown valves.

The calculated and measured dump line flow rates are compared in Figure 10. Initially there was no dump line flow. When the blowdown valves opened, the flow rate initially was very large. The flow then decreased along with the core pressure shown in Figure 7.

The calculated and measured pressurizer indicated levels are compared in Figure 11. Until the blowdown valves opened, the pressurizer level remained near its initial value. When the blowdown valves opened, the loss of primary system coolant caused a very rapid loss of the pressurizer level.

The calculated and measured primary coolant pump speeds are compared in Figure 12. The pump speed, initially 3200 rpm, declined rapidly when pump power was tripped at the beginning of the test. The exponential pump coastdown continued until the blowdown valves opened. At that time the primary system coolant upstream of the primary coolant pumps (in the hot legs, pressurizer, steam generator tubes, and pump suctions of the cold legs) was accelerated around the coolant loop and through the core toward the opened blowdown valves. This acceleration caused the momentary increase in the pump speed shown in Figure 12. Eventually, as these upstream regions voided, the acceleration effect diminished and the coastdown of the pumps resumed.

The different primary coolant pump coastdown characteristics shown in Figure 12 caused the different ECCS actuation times (48 s in the

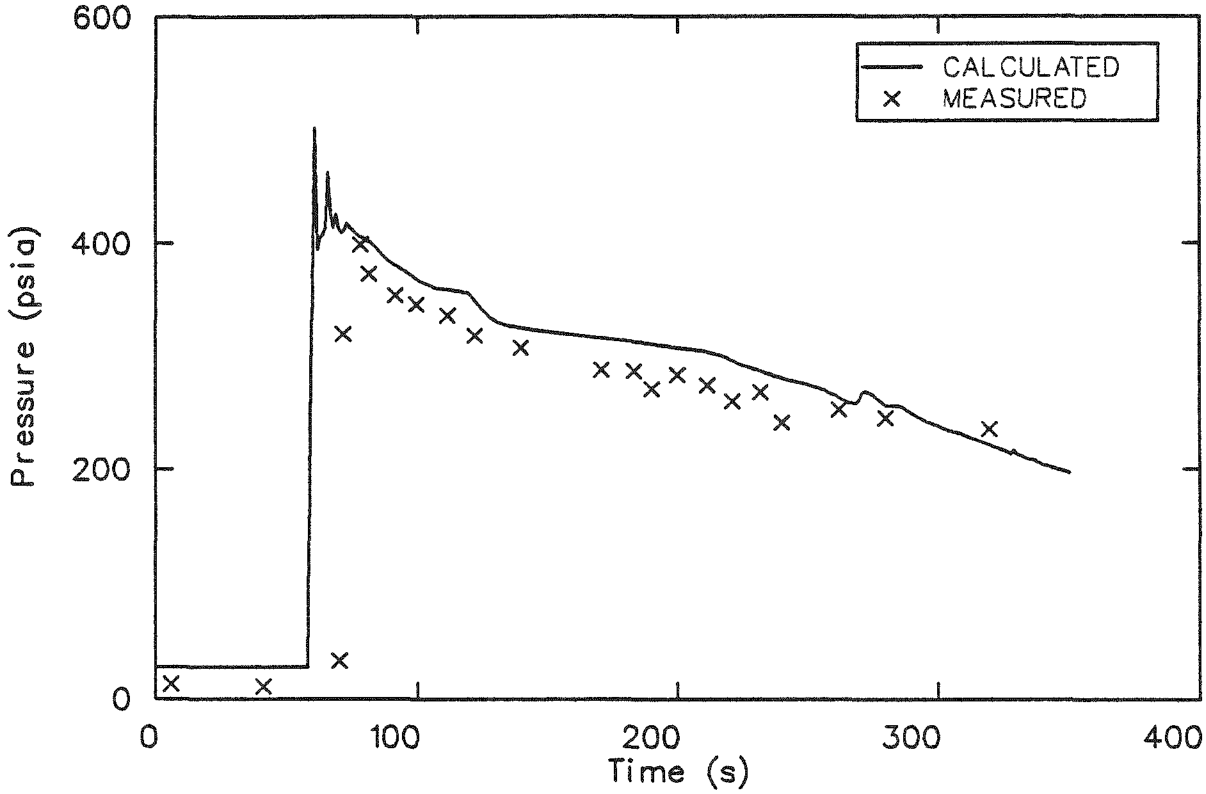

Figure 8. Measured and calculated pressures downstream ofblowdown valves for the hot dump test.

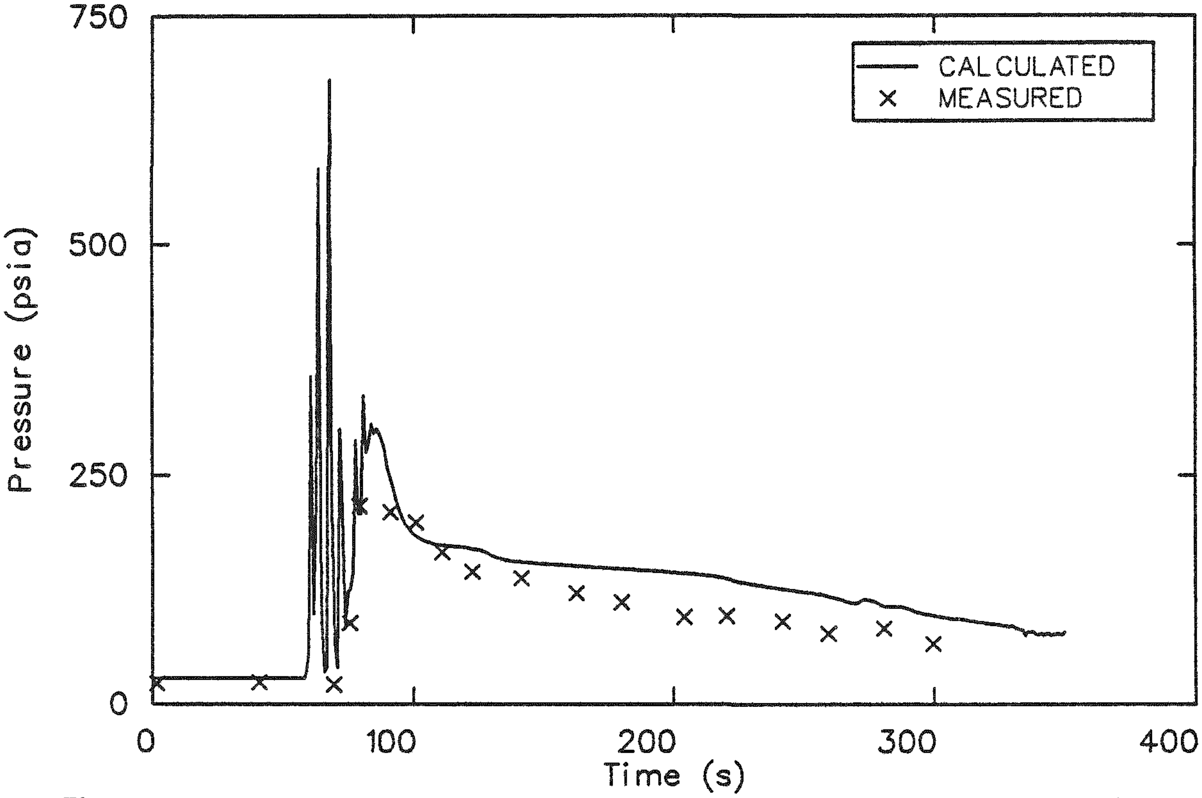

Figure 9. Measured and calculated pressures at dump basin sparger for the hot dump test.

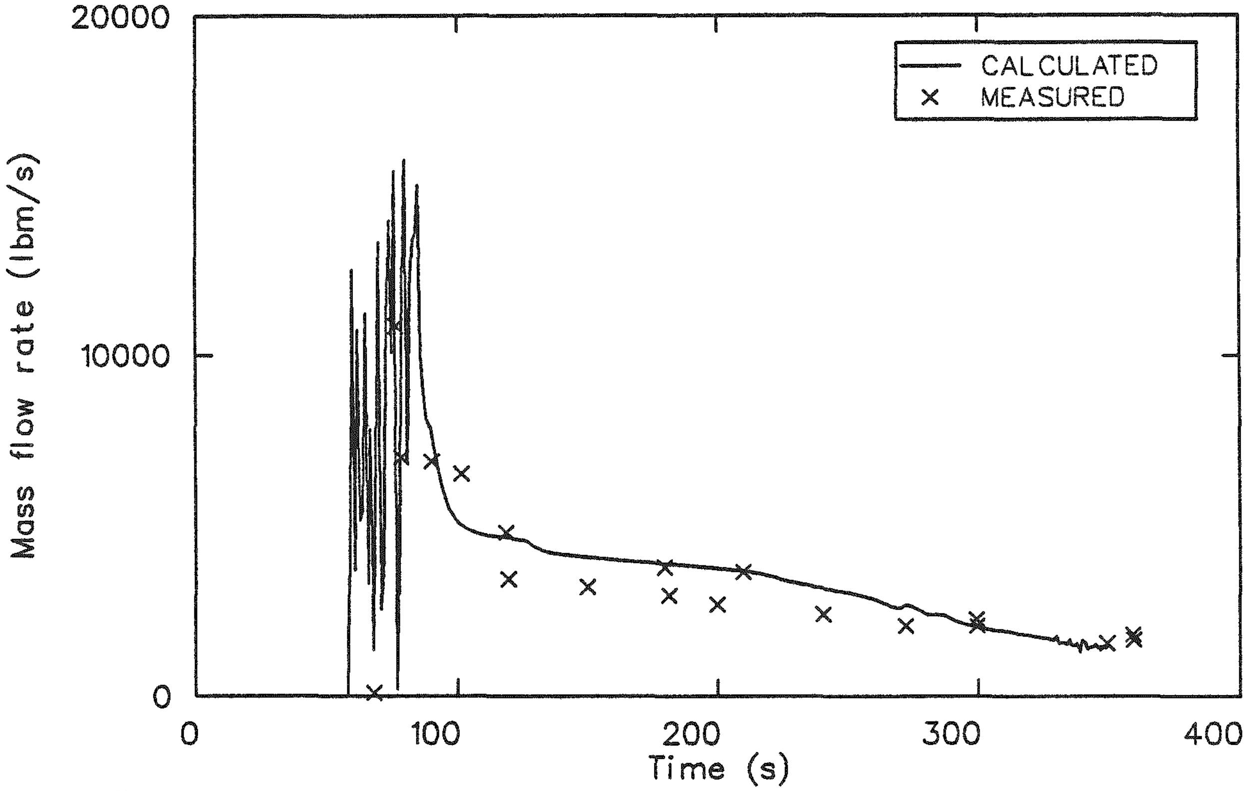

Figure 10. Measured and calculated dump line flow rateresponses for the hot dump test.

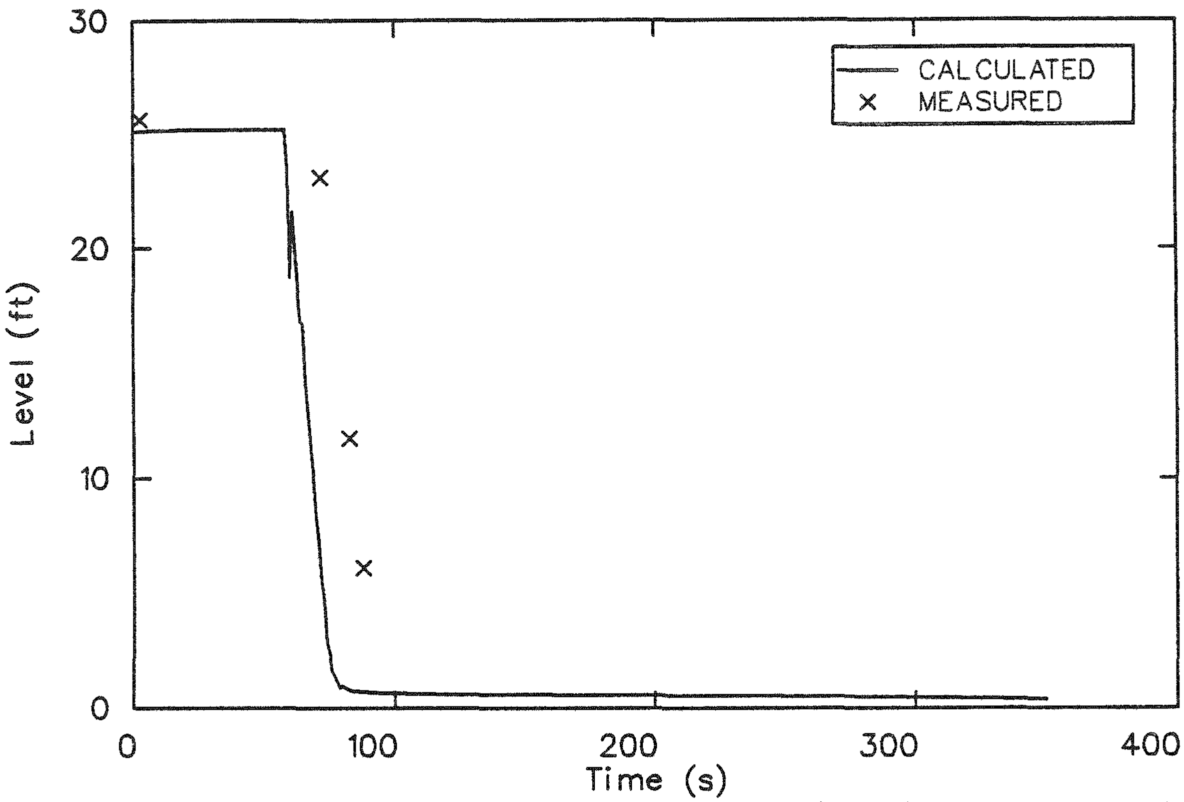

Figure 11. Measured and calculated pressurizer indicated level responses for the hot dump test.

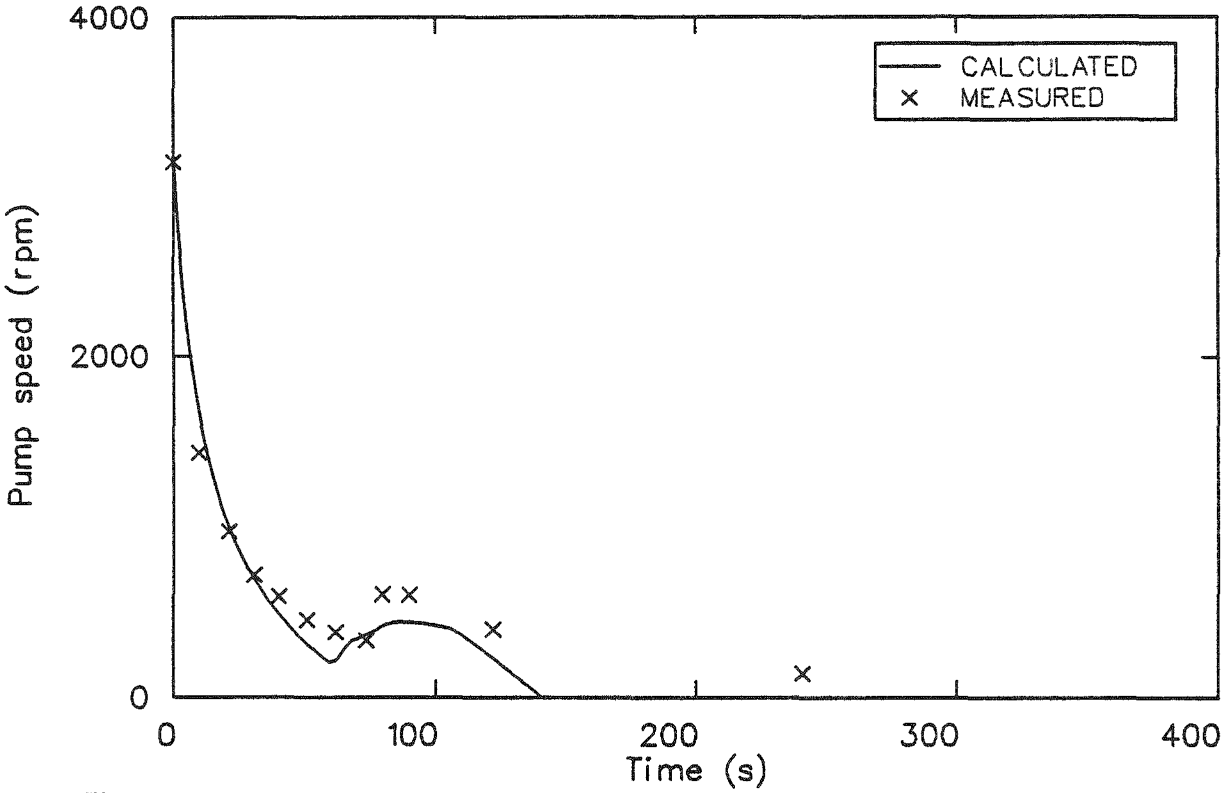

Figure 12. Measured and calculated primary coolant pump speeds for the hot dump test.

calculation, 70 s in the plant data). The faster pump coastdown in the calculation caused the low pump speed and low core flow rate criteria for ECCS actuation to be reached earlier. In the RELAP5 pump model, the coastdown is determined by the hydrodynamic forces on the pump rotor, the combined inertia of the pump motor and rotor, and the bearing friction. In the N Reactor model, a constant bearing friction equal to 3% of rated torque was assumed. The comparison of the measured and calculated pump coastdown suggests the modeled bearing friction is too large. A speed-dependent component of bearing friction is likely to be needed in the model to improve the comparison. During the momentary increase in pump speed from about 70 to 100 s the measured increase in speed was larger than that calculated, supporting the suggestion that modeled bearing friction is too large. Adjustment of the modeled bearing friction was considered but was not accomplished because: (1) the calculated and measured pump speeds were already in reasonable agreement, and (2) fully tripping the primary coolant pumps is not a frequent condition in N Reactor safety analysis. In the majority of N Reactor safety analyses, the reactor coolant pump speed is determined by a control system that simulates runback to a lower speed, and not total loss of pump motor power.

The differences in ECCS actuation time caused the offsets in blowdown timing observed in the pressure comparisons in Figures 7, 8, and 9. Disregarding these offsets, these three comparisons of calculated and measured pressures are very favorable. The calculated pressures at all three locations generally tracked well, and were only slightly above, the measured pressures.

With the calculated pressures slightly higher than the measured pressures, the calculated dump line mass flow rate was also slightly above the measured rate as shown in Figure 10.

The calculated rate at which the pressurizer level declined, (see Figure 11), also compared very well with the plant data. The pressurizer was empty at 80 s in the calculation and at 100 s in the measured plant data. This time to empty was 32 s and 30 s, respectively, after the blowdown valves opened. Therefore, the time required to empty the pressurizer following the opening of the blowdown valves was comparable in the calculated and measured data. The residual pressurizer level in the calculation indicates the head of steam residing between the pressurizer level tap elevations.

In summary, the comparisons of measured and RELAP5-calculated data for the hot dump test were very favorable. A difference in primary coolant pump coastdown characteristics, believed to be caused by an overstatement of the bearing friction in the computer model, led to minor differences in calculated and measured sequence event timing. Comparisons of measured and calculated pressures at several locations and dump line flow rate showed good agreement. These comparisons suggest the net modeled flow loss from the core to the dump basin is slightly high. Qualitatively, the results from these comparisons indicate the RELAP5/MOD2 computer code and the N Reactor model adequately simulate the thermal-hydraulic characteristics of the plant during two-phase blowdown transients.

6. CONCLUSIONS

Comparisons of calculated and measured plant data for the January 12, 1980 manual plant trip sequence indicated the RELAP5/MOD2 computer code and N Reactor model can adequately simulate the thermal-hydraulic phenomena for transients involving a single-phase primary coolant system.

Calculated and measured data for pressurizer pressure responses compared very well. The pressurizer level response comparison was adequate. The pressure and level comparisons indicated there likely is more lag in the plant pressurizer indicated level than has previously been thought. The HPI flow rate response did not compare well, but there is considerable uncertainty in the plant data for this parameter. These comparisons have provided a satisfactory global check on modeled system volumes, heat transfer areas, metal masses, flow characteristics, and control system processes.

Comparisons of calculated and measured plant data for the March 27, 1964 hot dump plant startup test indicated the RELAP5/MOD2 computer code and N Reactor model adequately simulate the thermal-hydraulic phenomena of the plant during two-phase blowdown transients.

Calculated and measured data for system pressures at several locations and dump line mass flow rate were in excellent agreement. A difference in primary coolant pump coastdown characteristics, believed to be caused by an overstatement of bearing friction in the model, led to minor differences in the calculated and measured event sequence timings.

7. REFERENCES

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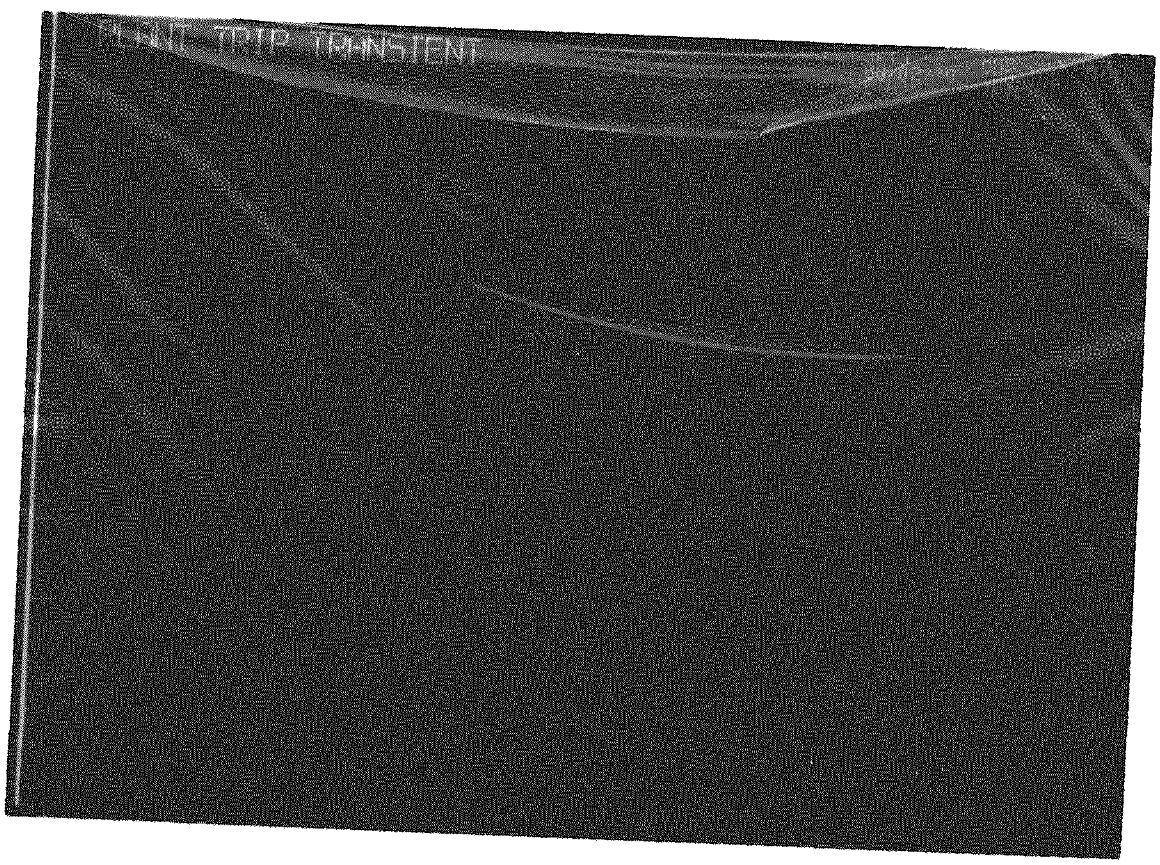
APPENDIX A

RELAP5 N REACTOR MODEL LISTING
FOR THE PLANT TRIP SEQUENCE

APPENDIX A

RELAP5 N REACTOR MODEL LISTING FOR THE PLANT TRIP SEQUENCE

The following is the input listing for the RELAP5 model of the N Reactor used for the plant trip calculation.



APPENDIX B

RELAP5 N REACTOR MODEL LISTING FOR THE HOT DUMP TEST SEQUENCE

APPENDIX B

RELAP5 N REACTOR MODEL LISTING FOR THE HOT DUMP TEST SEQUENCE

The following is the input listing for the RELAP5 model of the N Reactor used for the hot dump test calculation.