NRC FORM 474 (1-90) 10 CFR 55.45(b), 55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0138
EXPIRES: 9-30-92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 120 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0138), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

SIMULATION FACILITY CERTIFICATION

INSTRUCTIONS. This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information, and check the appropriate box to indicate reason for submittal.

FACILITY
Vermont Yankee Nuclear Power Station

DOCKET NUMBER
50- 271

LICENSEE
Vermont Yankee Nuclear Power Corporation

DATE
1/30/91

This is to certify that:
1. The above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR 55.45.
2. Documentation is available for NRC review in accordance with 10 CFR 55.45(b).
3. This simulation facility meets the guidance contained in ANSI/ANS 3.5, 1985, as endorsed by NRC Regulatory Guide 1.149.
   If there are any exceptions to the certification of this item, check here [ ] and describe fully on additional pages as necessary.

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY
Vermont Yankee Training Center
P.O. Box 169 Ferry Road
Brattleboro, Vermont 05301

[XX] SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. *(For performance tests conducted in the period ending with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING COMPLETED *(Attach additional page(s) as necessary, and identify the item description being continued.)*

- Computer Real Time Test
- Steady State & Normal Operation Test
- Transient Tests
- Malfunction Tests
- Simulator Operating Limits Test

For additional information see pp. 26 - 47 of this report

[XX] SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25% of performance tests per year for the four-year period commencing with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED. *(Attach additional page(s) as necessary, and identify the item description being continued.)*

- Performance Testing Schedule

For additional information see pp. 48 - 49 of this report

[ ] PERFORMANCE TESTING PLAN CHANGE. *(For any modification to a performance testing plan submitted on a previous certification.)*

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional page(s) as necessary, and identify the item description being continued.)*

Not Applicable

[ ] RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR § 55.45(b)(5)(iv). Attach additional page(s) as necessary, and identify the item description being continued.)*

Not Applicable

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE – AUTHORIZED REPRESENTATIVE	TITLE	DATE
Warren P. Murphy	Senior Vice President, Operations Vermont Yankee Nuclear Power Corp.	1-30-91

In accordance with 10 CFR § 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO: Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: One White Flint North
11555 Rockville Pike
Rockville, MD

NRC FORM 474 (1-90) 10 CFR 55.45(b), 55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION

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FACILITY: Vermont Yankee Nuclear Power Station

DOCKET NUMBER: 50 271

LICENSEE: Vermont Yankee Nuclear Power Corporation

DATE: 1/30/91

This is to certify that:

1. The above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR 55.45.
2. Documentation is available for NRC review in accordance with 10 CFR 55.45(b).
3. This simulation facility meets the guidance contained in ANSI/ANS 3.5, 1985, as endorsed by NRC Regulatory Guide 1.149.
   If there are any exceptions to the certification of this item, check here [ ] and describe fully on additional pages as necessary.

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY

Vermont Yankee Training Center
P.O. Box 169 Ferry Road
Brattleboro, Vermont 05301

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DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional page(s) as necessary, and identify the item description being continued)*

Not Applicable

[ ] RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR § 55.45(b)(5)(v). Attach additional page(s) as necessary, and identify the item description being continued.)*

Not Applicable

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PDR ADOCK 05000271
P PDR

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE – AUTHORIZED REPRESENTATIVE: Warren P Murphy

TITLE: Senior Vice President, Operations, Vermont Yankee Nuclear Power Corp.

DATE: 1-30-91

In accordance with 10 CFR § 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO: Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: One White Flint North, 11555 Rockville Pike, Rockville, MD

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR CERTIFICATION REPORT

PREPARED BY

VERMONT YANKEE TRAINING DEPARTMENT

SIMULATOR GROUP

FEBRUARY 1991

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REFERENCES

1. 10 CFR 55.45 (b) (5), Certification of Simulation Facilities
2. USNRC Regulatory Guide 1.149, Nuclear Power Plant Simulation Facility For Use In Operator License Examinations
3. ANSI/ANS - 3.5 - 1985, American National Standard Nuclear Power Plant Simulators For Use In Operator Training
4. Vermont Yankee Final Safety Analysis Report
5. Vermont Yankee Simulator Benchmarking Analysis Report
6. NUREG - 1259, Evaluation Procedure For Simulation Facilities Certified Under 10 CFR 55
7. Vermont Yankee Simulator Administration Procedures

VERMONT YANKEE SIMULATOR CERTIFICATION GENERAL INFORMATION

The Vermont Yankee simulator is a plant-referenced simulator owned and operated by Vermont Yankee Nuclear Power Corporation. The simulator was designed and built by the Singer Link Company of Columbia, Maryland, and was ready for training March 6, 1986.

The referenced plant is the Vermont Yankee Nuclear Power Station located in Vernon, Vermont. It is a 540 Mwe, General Electric Boiling Water Reactor, classified as a BWR 3/4. The plant went into operation in November 1972.

In accordance with 10 CFR 55.45 (b)(5) Vermont Yankee certifies that its simulator conforms to the commission's regulations as described in 10 CFR 55.45, USNRC Regulatory Guide 1.149, and ANSI/ANS - 3.5 - 1985. The format of this report reflects the guidance of Appendix A of reference (3) and as such includes the description of the simulator control room as compared to the station control room, the instructor console, the simulator design criteria, the simulator capabilities as tested, the simulator operating limits, and the simulator configuration management system.

VERMONT YANKEE SIMULATOR CERTIFICATION
DEGREE OF PANEL SIMULATION

CRP 9-2 This vertical back panel, AREA AND PROCESS MONITORING, is complete in its physical and functional aspects and is completely supported by the simulation software, hardware and the problem control module (PCM).

CRP 9-3 The right wing portion of the main bench board, REACTOR AND CONTAINMENT COOLING & ISOLATION, is complete in its physical and functional aspects and completely supported by the simulator software, hardware and PCM.

CRP 9-4 Right center portion of the main bench board, REACTOR WATER CLEAN-UP AND REACTOR RECIRCULATION, is complete in its physical and functional aspects and completely supported by the simulator software, hardware and PCM.

CRP 9-5 Center portion of the main bench boards, REACTOR CONTROL, is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-6 Left center portion of the main bench board, CIRCULATING COOLING WATER, AND FEEDWATER, is complete in its physical and functional aspects and supported by the simulation software, hardware and PCM.

CRP 9-7 Left portion of the Main Bench Board, TURBINE GENERATOR, is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-8 Extreme left portion of the main bench boards, ELECTRICAL AND DIESEL GENERATOR CONTROL, is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-10 This vertical back panel, PROCESS RADIATION MONITOR, is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-11 This vertical back panel, AREA RADIATION MONITOR, is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-12 This vertical back panel, NUCLEAR INSTRUMENTATION is complete in its physical and functional aspects and supported by the simulation software, hardware and PCM.

CRP 9-13 This vertical back panel, TRAVERSING IN-CORE PROBE (TIP), has only one TIP Channel Drawer that is functional. The original TIP Drive Control Monitor (112C3152) functions as per the specification. Pictures of the other two new drive control channels (945E258) are displayed in the proper locations of the drawers. The remainder of the panel is complete in its physical and functional aspects and supported by the simulator software, hardware and PCM.

CRP 9-14 The external hardware of the POWER RANGE NEUTRON MONITORING panel is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM. In addition, the APRM Channels B & E thumb wheel switches (20) internal to the panel along with the RBM cabinet FLOW CONVERTER UNITS are functional and supported by the simulator software, hardware and PCM.

CRP 9-15 The switches and lights on the vertical back panel, RPS 'A', are complete in their physical and functional aspects and supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind each relay bezel and they are not functional.

CRP 9-16 The individual ROD SCRAM SWITCHES on the vertical back panel are complete in their physical and functional aspects and are supported by the simulator software, hardware and PCM. All other hardware is a mock-up and is not functional.

CRP 9-17 The switches and lights on the vertical back panel, RPS 'B', are complete in their physical and functional aspects and supported by the simulator software, hardware and PCM. The relays are mock-ups with a photograph behind the relay bezel, and are not functional.

CRP 9-18 The front portion of this vertical back panel, FEEDWATER & RECIRCULATION, is a mock-up with the exception of the 11 adjustable controllers and one timer which are complete in their physical and functional aspects. They are supported by the simulation software, hardware and PCM.

CRP 9-19 The front portion of this vertical back panel, PROCESS INSTRUMENT, is a mock-up with the exception of the five controllers and four timers which are complete in their physical and functional aspects. They are supported by the simulation software, hardware and PCM.

CRP 9-20 The front portion of this vertical back panel, INSTRUMENTATION, is a complete mock-up. There is no functional application for this panel.

CRP 9-21 The vertical back panel, NUCLEAR STEAM TEMPERATURES, is complete in its physical and functional aspects and supported by the simulation software, hardware and PCM.

CRP 9-22 The vertical back panel, GENERATOR AND TRANSFORMER PROTECTION, lights and switches are complete in their physical and functional aspects and are supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind the bezel, and are not functional.

CRP 9-23 The vertical back panel, STEAM AND TURBINE AUXILIARIES, is complete in its physical and functional aspects and supported by the simulation software, hardware and PCM.

CRP 9-25 The vertical back panel, VENT AND DRYWELL, is complete in its physical and functional aspects and completely supported by the simulation software, hardware and PCM.

CRP 9-26 The vertical back panel, STANDBY GAS TREATMENT, upper portion meter, switches, and lights are complete in their physical and functional aspects and supported by the simulation software, hardware and PCM. The lower portion of the panel, converters and relays, are mock-ups and are not functional.

CRP 9-27 This vertical back panel, CONTROL ROD INFORMATION, has no physical or functional simulation aspects.

CRP 9-28 The vertical back panel, REACTOR MANUAL CONTROL, externally has no physical or functional simulation aspects. The interior consists of two switches and the ALTERNATE RWM. The interior I/O is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

CRP 9-30 The vertical back panel, RCIC LOGIC, has two switches that are simulated. They are physically and functionally supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind each relay bezel, and are not functional.

CRP 9-32 On the vertical back panel; RHR, CS, ADS LOGIC 'A', only the switches, lights and two active relays are functional and supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind each relay bezel, and are not functional.

CRP 9-33 On the vertical back panel; RHR, CS, ADS LOGIC 'B', only the switches, lights and two active relays are functional and supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind the relay bezel, and are not functional.

CRP 9-38	The vertical back panel, JET PUMP INSTRUMENTS, is a mock-up with the exception of the 20 Jet Pump meters which are physically and functionally supported by the simulation software, hardware, and PCM.
CRP 9-39	On the vertical back panel, HPCI LOGIC, the switches on the external portion of this panel are functionally supported by the simulation software, hardware and PCM. The relays are mock-ups with a photograph behind the bezel ring and are not functional.
CRP 9-41	The switches internal to this back panel, PCIS 1, are complete in their physical and functional aspects and supported by the simulator software, hardware and PCM. None of the internal relays are simulated.
CRP 9-42	The switches internal to this back panel, PCIS 2, are complete in their physical and functional aspects and supported by the simulator software, hardware and PCM. None of the internal relays are simulated.
CRP 9-45	The back panel, VITAL AC BREAKER, is complete in its physical and functional aspects and supported by the simulator software, hardware and PCM.
CRP 9-46	The INSTRUMENT AC BREAKER back panel is complete in its physical and functional aspects and supported by the simulator software, hardware and PCM.
CRP 9-47	This vertical back panel, CONTAINMENT ATMOSPHERE, is complete in its physical and functional aspects and is supported by the simulator software, hardware and PCM.
CRP 9-48	This vertical back panel, WEATHER RECORDER, is complete in its physical and functional aspects and is supported by the simulator software, hardware and PCM.
CRP 9-49	This vertical back end panel is a complete mock-up with photographs of the hardware behind each bezels, and they do not function.
CRP 9-50	The AOG vertical back panel is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.
TSIP	The right side end back panel, TURBINE SUPERVISORY, has two internal switches and one light that are supported by the simulation software, hardware and PCM.
RCIC ALT S/D PANEL 82-1	The RCIC Alt S/D panel is located in a separate room in the rear of the Simulator. It is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.

SRV A Alt S/D PANEL	This panel is attached to the RCIC Alt S/D panel. The single switch is functional and fully supported.
RHR ALT S/D PANEL CP 82-2	The RHR Alt S/D panel is located in a separate room in the rear of the Simulator. It is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM.
CAD A PANEL	The CAD 'A' MONITORING panel is located in the center rear of the simulator and is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM. The CAD 'B' panel is not simulated.
H_2/O_2 MONITOR PANEL	H_2/O_2 MONITORING panel is located near the CAD 'A' panel at the center rear of the Simulator and is complete in its physical and functional aspects and is supported by the simulation software, hardware and PCM. In the Control Room it is attached to the CAD panel. H_2/O_2 panel attached to the 'B' CAD panel is not simulated.
D/G ENGINE & LOCAL CONTROL PANEL	The DIESEL GENERATOR 'A' ENGINE and LOCAL CONTROL panel are located in a separate room in the rear of the simulator. These two panels are complete in their physical and functional aspects and are supported by the simulation software, hardware and PCM. The relays are mock-ups with photographs placed behind the relay bezels. The local panel is missing the stairs and lower half of the blank panel. D/G 'B' and its local panel are not simulated.
EMERGENCY FIRE DAMPER SWITCHES	The three EMERGENCY FIRE DAMPER SWITCHES are located near CRP 9-3 and are complete in their physical and functional aspects.
METEOROLOGICAL DECWRITER	Is located in the rear of the simulator. This output device is complete in its physical and functional aspects.
FIRE DETECTION PANEL	This panel is located in the rear of the simulator and is complete in it physical and functional aspects.

INSTRUCTOR STATION (PCM)	The PCM is located to the right as one enters the Simulator Room. The top of the PCM has one-way glass. It reflects the requirements of the simulator design specification.
CONTROL ROOM FURNITURE	The Control Room furniture (Shift Supervisor's desk, SCRO's desk, CRO's desk, ACRO's desk and the SE's desk) is replicated in the proper location in the Simulator Room.

The degree of panel simulation, as described above, has been reviewed by the simulation group and the Operations Training Supervisor. The assessment results indicate that no training value is lost due to the differences in fidelity.

PERFORMED BY: Allen Thomas, Gary LeClair, Al Chesley

VERMONT YANKEE SIMULATOR CERTIFICATION
CONTROL ROOM ENVIRONMENT

ARRANGEMENT:

The simulator and the control room are approximately the same size, 3600 sq. ft., but are proportioned differently. The control room has several side rooms to the left as you enter the main access door: A kitchen, toilet, operations offices (2) and a security office. With this area included, the control room is approximately 4150 sq. ft.

The dimensions of all the panels in the control room and the simulator are the same.

The CAD panel (the simulator only has a panel A) is located in the center rear of the simulator. The CAD panel in the control room is to the right rear of the room near the access doors. The H_2/O_2 Monitoring panel is attached to the CAD panel in the control room. The H_2/O_2 Monitoring panel is next to the CAD panel in the simulator.

The Fire panel is to the left rear of the simulator near CRP 9-49. The Fire panel in the control room is in the security office.

The area in the simulator between the right rear back panels and the rear wall is 4'5". This area in the control room is 10', which is the location of the Project Save Computer, PCIS Temperature Switch Test Panel, Security Computer, Seismic Monitor Workstation, CAD panels, and Used Strip Charts. The area to the left rear in the simulator is 7'2" from rear panel to wall. The same area in the control room is 4'.

The distances between the rows of panels are the same in both the control room and the simulator. The distance between the second and third row is 5', and between the third and forth row is 3'3". The distances between the front horseshoe and the second row, both left and right of the horseshoe is the same distance in both the control room and the simulator.

The distance from CRP 9-8 to the wall is 6'8" in the simulator and only 34" in the control room. The distance from CRP 9-3 and the wall is 6'2" in the simulator and only 46" in the control room.

LIGHTING AND CEILING:

The simulator has recessed fluorescent lighting arranged in metal acoustical ceiling tile squares. The Control Room has two sets of lighting: Half is a suspended ceiling with the fluorescent lighting distributed above the silver squared suspended ceiling tiles, the other half is a recessed fluorescent light in the normal acoustical rectangular tiles. The simulator has three emergency incandescent rectangular lights near the front panel horseshoe for Loss of Power scenario; the front fluorescent lights go out while the rear fluorescent lights for the back panel remain on. The control room has eight incandescent emergency lights for the front panels. Several back panels have similar lighting. The floor-to-ceiling height in the Simulator is 9'8" while the floor-to-ceiling height in the Control Room is 10'.

FLOORING:

The simulator is completely carpeted on a raised 2' x 2' computer floor. The color of the carpet is tan with the exception of approximately 4'6" area near the front panel, CRP 9-3 through CRP 9-8 which is a brown carpet. The Control Room is completely carpeted light grey between the front access doors and the desks. There is a multi-color grey carpet in the area from the front of the desks to the front panels and beyond in the walkways around CRP 9-3 and 9-8. The back panel area is completely carpeted in tan. The Control Room carpet is placed on a solid concrete floor.

ACCESS:

Normal access to the simulator is through the wooden door side entrance next to the CRP 9-8 (Unit #6) and the PCM. A second entrance is a glass door from the computer room to the simulator and is near CRP 9-3 (Unit #1). There is an entrance to the Training Center Electrical Room on the right rear side of the simulator.

Normal access to the control room is through the card-keyed double metal doors, directly behind the Shift Supervisor's desk. The other access door, also a card-keyed double metal door, is in the rear of the control room behind the TSIP Panel.

SIDE ROOMS:

There are three side rooms at the back of the simulator: two for the Alternate Shutdown panels, RCIC and RHR, and one for the two Diesel panels.

These rooms are not in the control room because the panels are located in their respective areas in the plant. The simulator also has a Hardware Support Area between the Alternate Shutdown Rooms and the Diesel Room.

OTHER DIFFERENCES:

A raised Instructor Station or Problem Control Monitor (PCM) is located inside the simulator to the right of the normal access.

The simulator has a viewing gallery to the right of the PCM and behind the Shift Supervisor's desk. It also has glass windows between the simulator and the Computer Room. All the glass walls have draw curtains to limit external observations as necessary. The control room has no glass walls or windows.

The size of the left operator's desk is smaller in the simulator than in the control room; this allows access between the desk and the PCM.

The communication systems, Gai-tronics and Emergency telephones are the same in both the simulator and the control room.

The simulator has a central vacuuming system with outlet in the floor in several locations throughout the simulator.

The chairs are different in the simulator than in the control room.

The HVAC Unit in the simulator is located behind CRP 9-5. Air vents are located in the floor in non-normal access areas. The exhaust is located in the ceiling. SAC-1 is the HVAC unit in the control room with the intake and exhaust located in the ceiling.

The simulator has a halon system that discharges from the ceiling. Smoke detectors are located in the ceiling, and below the flooring. Each control room panel has its own smoke detector. Some of the larger panels have several detectors.

The control room has an Emergency Breathing Air System, the simulator does not.

The location of the control room plant drawings and the simulator plant drawings are in different locations. Both are located near the Shift Supervisor's desk. The simulator has several cabinets near CRP 9-8 and 9-49, these cabinets contain the original Simulator Reference Data ase.

The panel door fixtures are different in the simulator than those in the control room.

The simulator has two cameras mounted in the ceiling and several ceiling mounted microphones that are used for recording. The control room does not have this feature.

The simulator has a digital clock used for ERFIS to show the differences in real and simulator run time. The control room does not have this digital clock on top of CRP 9-5 panel.

The simulator has a set of double doors in the left rear portion of the room for access to and from the warehouse area. These doors do not exist in the control room.

PERFORMED BY: ALLEN THOMAS

VERMONT YANKEE SIMULATOR CERTIFICATION
INITIAL CONDITION IDENTIFICATION

Note that these are the most recent ICs, and the initial conditions cited in the Transient test abstracts may be slightly different.

IC NO	RX PWR	CORE FLOW	RX PRESS (PSIG)	TEMP (F)	XE REACT	CORE LIFE	REMARKS
1	0	0	0	136	0	BOC	9/11/90 COLD S/D COND SYS IS DEPRESSURIZED
2	0	28	0	146	0	BOC	9/17/90 APPROACH TO CRITICAL SEQ 14-A1 GRP 2
3	1	28	50	295	0	BOC	9/17/90 HEATUP IN PROGRESS SEQ 14-A1 GRP 7
4	3	29	245	401	0	BOC	9/21/90 HEATUP IN PROGRESS SEQ 14-A1 GRP 13
5	4	30	887	528	1	BOC	9/21/90 STARTUP IN PROGRESS SEQ 14-A1 GRP 14
6	22	37	930	516	1	BOC	9/21/90 1 1/2 BYP ON EPR SEQ 14-A1 GRP 39
7	33	49	938	522	1	BOC	9/21/90 35% PWR SEQ 14-A1 GRP 52
8	71	59	972	516	104	BOC	9/21/90 70% PWR SEQ 14-A1 GRP 150
9	100	93	1007	525	104	BOC	9/14/90 100% PWR (SUMMER) SEQ 14-A1 GRP 150
10	99	93	1007	525	104	BOC	9/21/90 100% PWR (WINTER) SEQ 14-A1 GRP 150

VERMONT YANKEE SIMULATOR CERTIFICATION
INITIAL CONDITION IDENTIFICATION

IC NO	RX PWR	CORE FLOW	RX PRESS (PSIG)	TEMP (F)	XE REACT	CORE LIFE	REMARKS
11	0	28	1	214	0	MOC	9/13/90 APPROACH TO CRITICAL SEQ 14-A1 GRP 2
12	39	50	943	519	2	MOC	9/14/90 S/U IN PROGRESS SEQ 14-A1 GRP 66
13	71	57	970	515	107	MOC	9/13/90 70% PWR SEQ 14-A1 GRP 98
14	98	93	1003	525	105	MOC	9/13/90 100% PWR (SUMMER) SEQ 14-A1 GRP 98
15	100	94	1005	525	106	MOC	9/13/90 100% PWR (WINTER) SEQ 14-A1 GRP 98
16	0	28	2	218	0	EOC	9/13/90 APPROACH TO CRITICAL SEQ 14-B2 GRP 3
17	4	29	919	532	0	EOC	9/13/90 S/U IN PROGRESS SEQ 14-B2 GRP 14
18	71	63	972	518	106	EOC	9/13/90 70% PWR SEQ 14-B2 GRP 248
19	100	99	1008	526	105	EOC	9/13/90 100% PWR (WINTER) SEQ 14-B2 GRP 248
20	100	99	1008	527	105	EOC	9/13/90 100% PWR (SUMMER) SEQ 14-B2 GRP 248

VERMONT YANKEE SIMULATOR CERTIFICATION
SIMULATOR FEATURES

The simulator has several major capabilities which allow the instructor to dynamically control the training exercise. Some of these features are controlled through pushbutton indicators and others through the CRT displays. The following is a basic description of the major features. A more detailed description is contained in section II and III of the Instructor's Operations Manual.

INITIAL CONDITION

Allows the starting of a training session from a predetermined starting point.

FREEZE

Allows stopping and starting dynamic simulation.

SNAPSHOT

Allows the instructor to record the present status of the simulator for future use as an initial condition.

BACKTRACK

Allows the instructor to back up the simulation to a previously stored plant condition.

REPLAY RECORD I/O

The capability to record up to two hours of panel output for a later replay.

TIME CONTROL

Allows the simulator to run faster than or slower than real time.

MALFUNCTIONS

Abnormal operational conditions that can be inserted into the scenario to teach problem analysis and recovery procedures.

BOOLEAN TRIGGER

Allows the automatic triggering of malfunctions.

INSTRUCTOR DIRECTED ACTIONS

Allow the instructor to control equipment and systems external to the control room.

I/O OVERRIDE

Allows the instructor to set the state of any control room I/O device.

MONITORED PARAMETERS

Allow the instructor to monitor the values of all control room instrumentation.

ANALOG TREND

Allows up to eight analog parameters to trend concurrently on a brush recorder.

TRAINEE PERFORMANCE REVIEW

An automatic testing tool which monitors the students' performance and provides a printed output upon request.

COMPUTER AIDED EXERCISE PROGRAM

Allows the instructor to set up a predefined exercise that will run under computer control.

VERMONT YANKEE SIMULATOR CERTIFICATION
INSTRUCTOR DIRECTED ACTIONS

DIGITAL IDAS

CDR04	CST-86B SEAL WTR SPLY TO HOGG VL	CLOSE	OPEN
CDR08	CONDENSATE DEMIN A (SERVICE)	IN	OUT
CDR09	CONDENSATE DEMIN B (SERVICE)	IN	OUT
CDR10	CONDENSATE DEMIN C (SERVICE)	IN	OUT
CDR11	CONDENSATE DEMIN D (SERVICE)	IN	OUT
CDR12	CONDENSATE DEMIN E (SERVICE)	IN	OUT
CDR16	COND DEMIN SYS HI/LO BAL OVRIDE	NORM	RESET
CDR18	BYPS VLV OG-559 ARND AFTER CONDE	CLOSE	OPEN
CDR19	MAKE UP DEMIN SERVICE	IN	OUT
CSR05	CS-26A CS'A'FULL FLW TST VLV ACB	OPEN	CLOSE
CSR06	CS-26B CS'B'FULL FLW TST VLV ACB	OPEN	CLOSE
CSR07	CS-35A 'A' CS PRESS LINE ISO VLV	CLOSE	OPEN
CSR08	CS-35B 'B' CS PRESS LINE ISO VLV	CLOSE	OPEN
CSR09	'A' CORE SPRAY PUMP BRKR	OPEN	CLOSE
CSR10	'B' CORE SPRAY PUMP BRKR	OPEN	CLOSE
CSR11	CS-5A 'A' CS MIN FLW VLV ACB	OPEN	CLOSE
CSR12	CS-5B 'B' CS MIN FLW VLV ACB	OPEN	CLOSE
CSR13	CS-7A 'A' CS TORUS SUCT VLV ACB	OPEN	CLOSE
CSR14	CS-7B 'B' CS TORUS SUCT VLV ACB	OPEN	CLOSE
CUR01	CUFD 'A' RWCU DEMINERALIZER	INSERV	OUTSE
CUR02	CUFD 'B' RWCU DEMINERALIZER	INSERV	OUTSE
CUR03	CU-63 CLNUP&CRD DIS MN ISO V FWL	CLOSE	OPEN
CUR06	RX WATER CLEARUP ANNUNCIATOR PANEL	NORM	RESET
DGR01	'B' DIESEL LOCKOUT RELAY	TRIP	RESET
DGR04	'A'DG FUEL RACK MECH LATCH(& TR	RESET	TRIP
DGR05	1A D/G ALT SHDN XFER SW SS-33/	NORM	EMERG
DGR06	1A D/G BKR SWITCH(LOCAL KEYLOCK)	TRIPNORMCLSE	
DGR07	'B' D/G ANNUNCIATOR PANEL	NORM	RESET
EDR01	125V DC CNTL PWR TO 4KV TO BUS 1	NORM	ALTER
EDR02	125V DC CNTL PWR TO 4KV TO BUS 2	NORM	ALTER
EDR03	125V DC CNTL PWR TO 4KV TO BUS 3	NORM	ALTER
EDR04	BUS 8 TIE TO MCC68	OPEN	CLOSE
EDR05	UPS MNTEN TIE BKR MCC889A MCC9B	OPEN	CLOSE
EDR06	UPS MNTEN TIE BKR MCC889B MCC8B	OPEN	CLOSE
EDR07	UPS ALT SUPLY BKR AT MCC89A	OPEN	CLOSE
EDR08	UPS ALT SUPLY BKR AT MCC89B	OPEN	CLOSE
EDR09	UPS SUPLY BKR AT 480V BUS 8	OPEN	CLOSE
EDR10	UPS SUPLY BKR AT 480V BUS 9	OPEN	CLOSE
EDR11	UPS OUTPUT BKR AT MCC89A	OPEN	CLOSE
EDR12	UPS OUTPUT BKR AT MCC89B	OPEN	CLOSE
EDR13	VITAL MG DC INPUT BKR	OPEN	CLOSE
EDR14	VITAL MG AC INPUT BKR	OPEN	CLOSE
EDR15	VITAL MG START SWITCH	OFF AUTRDCRN	
EDR16	VITAL MG AC OUTPUT BKR	OPEN	CLOSE
EDR17	125 VDC BUS DC-3A MN THRVER SWCH	NORM	EMERG
EDR18	125 VDC BUS DC-2A MN THRVER SWCH	NORM	EMERG
EDR19	125 VDC BUS DC-3 MN THRVER SWCH	NORM	EMERG
EDR20	120/240 VAC INSTR BS AUTOXFER SW	NORM	EMERG
EDR21	480 VAC BUS 8 LNP LD SHD RLY(1LD	NORM	RESET

EDR22	480 VAC MCC8A LNP LD SHD RLY(5LD	NORM	RESET
EDR23	480 VAC MCC8B LNP LD SHD RLY(5LD	NORM	RESET
EDR24	480 VAC MCC8C LNP LD SHD RLY(3LD	NORM	RESET
EDR25	480 VAC MCC9A LNP LD SHD RLY(3LD	NORM	RESET
EDR26	480 VAC MCC9B LNP LD SHD RLY(6LD	NORM	RESET
EDR27	480 VAC MCC9C LNP LD SHD RLY(1LD	NORM	RESET
EDR28	125 VDC PN DC-2-AS 'A'D/G DC SPL	OPEN	CLOSE
EDR29	125VDC DC-2-AS ALT PWR 4KV BUS 4	OPEN	CLOSE
EDR30	125VDC DC-2-AS ALT PWR 480KV B 4	OPEN	CLOSE
EDR31	125VDC DC-2-AS EMG DC FD MTS13-2	OPEN	CLOSE
EDR32	480V BUS 9 ALT SHUTDN X-FER SWIT	NORM	EMERG
EDR33	480V BUS 9 EMGCN PWR XFER SWITCH	NORM	EMERG
EDR34	BKR 49 ALT SHDN XFER SW SS-330	NORM	EMERG
EDR35	4KV BUS EMG CNTL PWR XFER SW	NORM	EMERG
EDR36	125VDC DC-1-AS DC FD TO CP-82-1	OPEN	CLOSE
EDR37	125VDC DC-1-AS EMG FD MTS-13-1	OPEN	CLOSE
EDR38	MTS-13-1 XFER SW EMG PWR CP-82-1	NORM	EMERG
EDR40	DELETED		
EDR41	4KV BKR 49 CROSSTIE (LOC KEYLOCK	TRIPNORMCLSE	
EDR42	JDDG START AND TIE TO MCC 9	OPEN	CLOSE
EGR02	MAIN GEN CORE MONITOR	IN	OUT
EGR03	MAIN GENER DISCONNECT LINKS	INSTAL	REMOVE
EGR04	STAT CLG PM RES PM TST 3WY VSC45	NORM	TEST
EVR01	SEAS SEL-AIR/WTR DEG F	A B C	
FPR01	DIESEL FIRE PUMP	HANDAUTOOFF	
FPR02	ELECTRIC FIRE PUMP	HANDAUTOOFF	
FWR06	CST-86B SEAL WTR SPLY TO HOGG VL	CLOSE	OPEN
FWR10	CONDENSATE DEMIN A (SERVICE)	IN	OUT
FWR11	CONDENSATE DEMIN B (SERVICE)	IN	OUT
FWR12	CONDENSATE DEMIN C (SERVICE)	IN	OUT
FWR13	CONDENSATE DEMIN D (SERVICE)	IN	OUT
FWR14	CONDENSATE DEMIN E (SERVICE)	IN	OUT
FWR18	COND DEMIN SYS HI/LO BAL OVRIDE	NORM	RESET
FWR20	BYPS VLV OG-559 ARND AFTER CONDE	CLOSE	OPEN
FWR21	MAKE UP DEMIN SERVICE	IN	OUT
FWR22	CONDENSATE DEMIN ANNUNCIATOR PANEL	NORM	RESET
HPR01	HPCI-25 ACB MIN FLOW VALVE	OPEN	CLOSE
IAR[illegible]	SA-82A SERV AIR SPLY TO'A'D/G VL	CLOSE	OPEN
IA[illegible]	SA-82B SERV AIR SPLY TO'B'D/G VL	CLOSE	OPEN
IA[illegible]	'A' AIR COMPRESSOR SWITCH	LEADOFF	LAG
IAR[illegible]	'B' AIR COMPRESSOR SWITCH	LEADOFF	LAG
IAR05	'C' AIR COMPRESSOR SWITCH	LEADOFF	LAG
IAR06	'D' AIR COMPRESSOR SWITCH	LEADOFF	LAG
IAR07	CONTAINMENT AIR COMPRESSOR SW	HANDOFF	AUTO
IAR08	IA-90D IA TO CONT X-CONN(SPEC FL	CLOSE	OPEN
IAR09	IA-73/73A/ SPECTACLE FLANGE	CLOSE	OPEN
IAR10	'A' AIR COMPRESSOR SWITCH	RUN	EMERST
IAR11	'B' AIR COMPRESSOR SWITCH	RUN	EMERST
IAR12	'C' AIR COMPRESSOR SWITCH	RUN	EMERST
IAR13	'D' AIR COMPRESSOR SWITCH	RUN	EMERST
IAR14	'A' INSTR DRYWER ISOLATION	NORM	ISOLAT
IAR15	AIR COMPRESSORS HIGH TEMP MANU RESET	NORM	RESET
MCR01	C-45A AIR EJECTOR 1ST STG DRN VL	CLOSE	OPEN
MCR02	C-45B AIR EJECTOR 1ST STG DRN VL	CLOSE	OPEN
MCR03	CND WTR BOX PRIMING PM P52-1A+1B	BRASOFF	ARBS
MSR01	AS-1 MAIN STEAM TO SJAE VLV	CLOSE	OPEN

NMR01	APRM 'B', LPRM INPUT	2B-16-33	OPER	BYPASS
NMR02	APRM 'B', LPRM INPUT	3B-16-25	OPER	BYPASS
NMR03	APRM 'B', LPRM INPUT	4B-32-17	OPER	BYPASS
NMR04	APRM 'B', LPRM INPUT	5B-32-09	OPER	BYPASS
NMR05	APRM 'B', LPRM INPUT	6B-24-09A	OPER	BYPASS
NMR06	APRM 'B', LPRM INPUT	1D-16-41	OPER	BYPASS
NMR07	APRM 'B', LPRM INPUT	2D-32-33	OPER	BYPASS
NMR08	APRM 'B', LPRM INPUT	4D-32-25	OPER	BYPASS
NMR09	APRM 'B', LPRM INPUT	5D-16-17	OPER	BYPASS
NMR10	APRM 'B', LPRM INPUT	6D-16-09	OPER	BYPASS
NMR11	APRM 'B', LPRM INPUT	1A-24-41	OPER	BYPASS
NMR12	APRM 'B', LPRM INPUT	2A-08-33	OPER	BYPASS
NMR13	APRM 'B', LPRM INPUT	4A-08-25	OPER	BYPASS
NMR14	APRM 'B', LPRM INPUT	5A-40-25	OPER	BYPASS
NMR15	APRM 'B', LPRM INPUT	6A-24-17	OPER	BYPASS
NMR16	APRM 'B', LPRM INPUT	1C-24-33	OPER	BYPASS
NMR17	APRM 'B', LPRM INPUT	2C-24-25	OPER	BYPASS
NMR18	APRM 'B', LPRM INPUT	3C-08-17	OPER	BYPASS
NMR19	APRM 'B', LPRM INPUT	4C-40-17	OPER	BYPASS
NMR20	APRM 'B', LPRM INPUT	5C-08-09	OPER	BYPASS
NMR21	APRM 'C', LPRM INPUT	2B-32-33	OPER	BYPASS
NMR22	APRM 'C', LPRM INPUT	5B-16-17	OPER	BYPASS
NMR23	APRM 'C', LPRM INPUT	2D-16-33	OPER	BYPASS
NMR24	APRM 'C', LPRM INPUT	4D-32-17	OPER	BYPASS
NMR25	APRM 'C', LPRM INPUT	6D-24-09C	OPER	BYPASS
NMR26	APRM 'C', LPRM INPUT	2A-24-25	OPER	BYPASS
NMR27	APRM 'C', LPRM INPUT	5A-08-09	OPER	BYPASS
NMR28	APRM 'C', LPRM INPUT	1C-24-41	OPER	BYPASS
NMR29	APRM 'C', LPRM INPUT	4C-08-25	OPER	BYPASS
NMR30	APRM 'C', LPRM INPUT	5C-40-25	OPER	BYPASS
NMR31	APRM 'E', LPRM INPUT	1B-24-33	OPER	BYPASS
NMR32	APRM 'E', LPRM INPUT	2B-24-25	OPER	BYPASS
NMR33	APRM 'E', LPRM INPUT	3B-08-17	OPER	BYPASS
NMR34	APRM 'E', LPRM INPUT	4B-40-17	OPER	BYPASS
NMR35	APRM 'E', LPRM INPUT	5B-08-09	OPER	BYPASS
NMR36	APRM 'E', LPRM INPUT	1D-24-41	OPER	BYPASS
NMR37	APRM 'E', LPRM INPUT	2D-08-33	OPER	BYPASS
NMR38	APRM 'E', LPRM INPUT	4D-08-25	OPER	BYPASS
NMR39	APRM 'E', LPRM INPUT	5D-40-25	OPER	BYPASS
NMR40	APRM 'E', LPRM INPUT	6D-24-17	OPER	BYPASS
NMR41	APRM 'E', LPRM INPUT	2A-16-33	OPER	BYPASS
NMR42	APRM 'E', LPRM INPUT	3A-16-25	OPER	BYPASS
NMR43	APRM 'E', LPRM INPUT	4A-32-17	OPER	BYPASS
NMR44	APRM 'E', LPRM INPUT	5A-32-09	OPER	BYPASS
NMR45	APRM 'E', LPRM INPUT	6A-24-09D	OPER	BYPASS
NMR46	APRM 'E', LPRM INPUT	1C-16-41	OPER	BYPASS
NMR47	APRM 'E', LPRM INPUT	2C-32-33	OPER	BYPASS
NMR48	APRM 'E', LPRM INPUT	4C-32-25	OPER	BYPASS
NMR49	APRM 'E', LPRM INPUT	5C-16-17	OPER	BYPASS
NMR50	APRM 'E', LPRM INPUT	6C-16-09	OPER	BYPASS
NMR51	APRM 'F', LPRM INPUT	1B-16-41	OPER	BYPASS
NMR52	APRM 'F', LPRM INPUT	4B-32-25	OPER	BYPASS
NMR53	APRM 'F', LPRM INPUT	6B-16-09	OPER	BYPASS
NMR54	APRM 'F', LPRM INPUT	3D-16-25	OPER	BYPASS
NMR55	APRM 'F', LPRM INPUT	5D-32-09	OPER	BYPASS
NMR56	APRM 'F', LPRM INPUT	1A-24-33	OPER	BYPASS

NMR57	APRM 'F', LPRM INPUT 3A-08-17	OPER	BYPASS
NMR58	APRM 'F', LPRM INPUT 4A-40-17	OPER	BYPASS
NMR59	APRM 'F', LPRM INPUT 2C-08-33	OPER	BYPASS
NMR60	APRM 'F', LPRM INPUT 6C-24-17	OPER	BYPASS
NMR79	TIP MACHINE SELECT	1 2	3
OGR02	AOGCCW 'A' COOLING FAN CNTL SW	OFF	AUTO
OGR03	AOGCCW 'B' COOLING FAN CNTL SW	OFF	AUTO
OGR04	AOGCCW 'A' COOLING PUMP CNTL SW	OFF	ON
OGR05	AOGCCW 'B' COOLING PUMP CNTL SW	OFF	ON
OGR09	AOG VACCUM DRAG VALVE OG-833A	OPEN	CLOSE
PCR01	V-16-19-5A DRWEL/TORUS VAC BKR	NORM	TEST
PCR03	LOCAL SWITCH SW-2 OUTSIDE DG ROOM TEF-2	AUTO	RUN
PCR04	RX BLG VENTLN SUP FAN (RSF-1A)	OFF AUTOON	
PCR05	RX BLG VENTLN SUP FAN (RSF-1B)	OFF AUTOON	
PCR06	RX BLG VENTLN SUP FAN SELETR SW	ASTBOFF BSTB	
PCR07	RX BLG VENTLN EXHST FAN (REF-1A	OFF NORMON	
PCR08	RX BLG VENTLN EXHST FAN (REF-1B	OFF NORMON	
PCR09	RX BLG VENTLN EXHST FAN SEL SW1	ASTBOFF BSTB	
PCR10	TURB BLG VENTLN SUP FAN (TSF-1A	OFF AUTOON	
PCR11	TURB BLG VENTLN SUP FAN (TSF-1B	OFF AUTOON	
PCR12	TURB BLG VENTLN SUP FAN SEL SW1	ASTBOFF BSTB	
PCR13	TURB BLG VENTLN SUP FAN (TSF-2A	OFF AUTOON	
PCR14	TURB BLG VENTLN SUP FAN (TSF-2B	OFF AUTOON	
PCR15	TURB BLG VENTLN SUP SELECTOR SW	ASTBOFF BSTB	
PCR16	TURB BLG VENTLN EXHST FAN (TEF1A)	OFF NORMON	
PCR17	TURB BLG VENTLN EXHST FAN (TEF1B)	OFF NORMON	
PCR18	TURB BLG VENTLN EXHST FAN SEL S	ASTBOFF BSTB	
PCR19	6INCH NITROGEN SPECTACLE INSTLD	NOFLOW FLOW	
PCR20	HVAC ANNUNCIATOR PANEL	NORM	RESET
PCR21	NITROGEN LINE (2 IN AC-29) SPEC FLNG	FLOW	NOFLOW
PCR22	RRU-5 RHRSW PUMP AREA SW-23	OFF AUTORUN	
PCR23	RRU-6 RHRSW PUMP AREA SW-24	OFF AUTORUN	
PCR24	RRU-7 RHR & CS PUMP AREA SW-25	OFF AUTORUN	
PCR25	RRU-8 RHR & CS PUMP AREA SW-26	OFF AUTORUN	
PPR02	PRINTER COMPUTER ROOM	OFF	ON
PPR03	PRINTERS SIMULATOR ROOM	OFF	ON
PPR04	LIMIT CYCLE OSCILLATIONS	NO	YES
RCR01	RCIC-27 MINIMUM FLOW VLV ACB	OPEN	CLOSE
RCR03	RCIC TRIP & THROTTLE VLV	NORM	RESET
RCR04	RCIC-V13-16 STM SPLY LN ISO VL	CLOSE	OPEN
RCR05	RCIC-V13-15 STM SPLY LN ISO VL	CLOSE	OPEN
RCR06	CP-82-3 RCIC ALT SHDN XFR S-1188	NORM	EMERG
RCR07	CP-82-3 RCIC ALT SHDN XFR S-1189	NORM	EMERG
RCR08	MTS-13-2 MAN THVR EM PWR RCIC-16	NORM	EMERG
RCR09	RCIC V13-15 STM SPLY LN ISO VL88	CLSENORMOPEN	
RCR10	RCIC V13-16 STM SPLY LN ISO VL89	CLSENORMOPEN	
RDR01	CRD-94 STATION DISCH TO RX VLV	CLOSE	OPEN
RDR06	CRD DR WTR FILTR F-16-A (SERVICC	IN	OUT
RDR07	CRD DR WTR FILTR F-16-B (SERVICC	IN	OUT
RDR08	CRD FLW CNTL STATION FCV-19A(SER	IN	OUT
RDR09	CRD FLW CNTL STATION FCV-19B(SER	IN	OUT
RDR10	REFUELING INTERLOCKS (ROD BLKS)	OFF	ON
RHR03	RHR-16A 'A'LOOP MIN FLW VL ACB	OPEN	CLOSE
RHR04	RHR-16B 'B'LOOP MIN FLW VL ACB	OPEN	CLOSE
RHR07	RHR-66 (DRN)RHR TO RADWAS VL ACB	OPEN	CLOSE
RHR08	RHR-26A RHR CONT SPR ISO VL ACB	OPEN	CLOSE

RHR09	RHR-26B RHR CONT SPR ISO VL ACB	OPEN	CLOSE
RHR10	RHR-38A TORUS SPR ISO VLV ACB	OPEN	CLOSE
RHR11	RHR PUMP 'A' ACB	OPEN	CLOSE
RHR12	RHR PUMP 'B' ACB	OPEN	CLOSE
RHR13	RHR PUMP 'C' ACB	OPEN	CLOSE
RHR14	RHR PUMP 'D' ACB	OPEN	CLOSE
RHR15	RHR-34A CONT SPR TST BYPS VL ACB	OPEN	CLOSE
RHR16	RHR-34B CONT SPR TST BYPS VL ACB	OPEN	CLOSE
RHR17	RHR-39A CONT SPR TST DISCH ACB	OPEN	CLOSE
RHR18	RHR-39B CONT SPR TST DISCH ACB	OPEN	CLOSE
RHR19	RHR ROOM WTR TIGHT DOOR - NE	OPEN	CLOSE
RHR20	RHR ROOM WTR TIGHT DOOR - SE	OPEN	CLOSE
RHR37	RHR-17/18 SHDW CLG SU ISO VL LCO SW	LOCK	OPEN
RHR39	1A RHR PM ALT SHDN XFER SW 1031	NORM	EMERG
RHR40	RHR-17 ALT PWR BKR (ON BUS 9)	OPEN	CLOSE
RHR41	A RHR PUMP SWITCH(LOCAL KEYLOK)	TRIPNORMCLSE	
RPR01	'A' RPS MG SET OUTPUT BKR	OPEN	CLOSE
RPR02	'B' RPS MG SET OUTPUT BKR	OPEN	CLOSE
RPR03	RPS ALT SPLY BKR CB1 480V MCC 8B	OPEN	CLOSE
RPR04	SRM 'A' SHORTING LINK	REMOVE	INSTAL
RPR05	SRM 'B' SHORTING LINK	REMOVE	INSTAL
RPR06	SRM 'C' SHORTING LINK	REMOVE	INSTAL
RPR07	SRM 'D' SHORTING LINK	REMOVE	INSTAL
RPR08	RPS CH A3 SHORTING LINKS	REMOVE	INSTAL
RPR09	RPS CH B3 SHORTING LINKS	REMOVE	INSTAL
RPR10	RPS APRM BUS A RESET SWITCH	NORMAL	RESET
RPR11	RPS APRM BUS B RESET SWITCH	NORMAL	RESET
RPR12	MSL LO LV HI FLO NOT IN RUN BYPASS	NORMAL	BYPASS
RPR13	RCU LO LVL/HI TEMP ISOL BYPASS	NORMAL	BYPASS
RPR14	RCU LO LVL ISOLATION BYPASS	NORMAL	BYPASS
RPR15	HPCI CST RETURN INTERLOCK BYPASS	NORMAL	BYPASS
RPR16	RCIC CST RETURN INTERLOCK BYPASS	NORMAL	BYPASS
RPR17	RB VENT ISOLATION BYPASS	NORMAL	BYPASS
RPR18	SLC TNK LINEUP TO RX VSL	NORMAL	TSTTNK
RPR19	RX HI LVL RFP TRIP BYPASS	NORMAL	BYPASS
RRR01	'A' RECIRC M.G LOCKOUT RELAY	NORM	RESET
RRR02	'B' RECIRC M.G LOCKOUT RELAY	NORM	RESET
RRR03	'A' RECIRC M.G D.C LUBE OIL PMP	STOPNORMSTRT	
RRR04	'B' RECIRC M.G D.C LUBE OIL PMP	STOPNORMSTRT	
RRR05	'A' RECIRC M.G SCOOP TUBE POWER	ON	OFF
RRR06	'B' RECIRC M.G SCOOP TUBE POWER	ON	OFF
RRR09	INSTRUMENT ROOT VALUE 14A/15A	CLOSE	OPEN
RRR10	INSTRUMENT ROOT VALUE 14B/15B	CLOSE	OPEN
RRR11	BUS 89A SPLY BKR V2-53A REC PM	CLOSE	OPEN
RRR12	BUS 89B SPLY BKR V2-53B REC PM	CLOSE	OPEN
RRR13	BUS 89A SPLY BKR V2-43A REC PM	CLOSE	OPEN
RRR14	BUS 89B SPLY BKR V2-43B REC PM	CLOSE	OPEN
RWR01	RWM TRANSFER SELECTION	GEPAC	ERFIS
SLR01	SLC-20/DW-41 SBLC TANK MAKEUP	CLOSE	OPEN
SLR02	SLC-15 MAN ISO COMB DIS ST O DRW	CLOSE	OPEN
SLR03	SLC-12A SLC PM A MAN ISO SUC VLV	CLOSE	OPEN
SWR01	RCW-32A RBCCW CLG TO RDWS,A&C PM	CLOSE	OPEN
SWR02	RCW-24A RBCCW CLG TO CRD B&D PMP	CLOSE	OPEN
SWR03	RCW-28A RBCCW RTN RHR CRD RW HDR	CLOSE	OPEN
SWR04	SW-32B ALT CLG RDWS,A,C RHR PMPS	CLOSE	OPEN
SWR05	SW-24B ALT CLG TO CRD,B,D RHR PM	CLOSE	OPEN

SWR06	RCW-42A RHR SW X-TIE TO D/G	CLOSE	OPEN
SWR07	RCW-42B RHR SW X-TIE TO D/G	CLOSE	OPEN
SWR09	SW-16B ALT CLG SPLY RHR SW PMPS	CLOSE	OPEN
SWR10	SW-17/SW-11 AH CLG RTN RH SW2 CT	CLOSE	OPEN
SWR11	SW-36A SW LOOP 'A' X-TIE ALT CLG	CLOSE	OPEN
SWR12	SW-36B SW LOOP 'B' X-TIE ALT CLG	CLOSE	OPEN
SWR17	RCW-29A/SW-29 MAN ISO VL RBCCW	CLOSE	OPEN
SWR18	SW-92A/101 SW IN/OUT MAN ISO V A	CLOSE	OPEN
SWR19	SW-92B/90 SW IN/OUT MAN ISO VL B	CLOSE	OPEN
SWR20	TCW-60A/61A TBCCW HX'A' MA IS VL	CLOSE	OPEN
SWR21	TCW-60B/61B TBCCW HX'B' MA IS VL	CLOSE	OPEN
SWR22	SW-62A/TCV-3 SW TO-FR TBCCW HX A	CLOSE	OPEN
SWR23	SW-62B/TCV-6 SW TO-FR TBCCW HX B	CLOSE	OPEN
SWR24	RCW-93A/93B RBCCW HX 'A'MA IS VL	CLOSE	OPEN
SWR25	RCW-91A/91B RBCCW HX 'B'MA IS VL	CLOSE	OPEN
SWR26	RHR-SW PUMP 'A' BREAKER	OPEN	CLOSE
SWR27	RHR-SW PUMP 'B' BREAKER	OPEN	CLOSE
SWR28	RHR-SW PUMP 'C' BREAKER	OPEN	CLOSE
SWR29	RHR-SW PUMP 'D' BREAKER	OPEN	CLOSE
SWR30	SW PUMP 'A' BREAKER	OPEN	CLOSE
SWR31	SW PUMP 'B' BREAKER	OPEN	CLOSE
SWR32	SW PUMP 'C' BREAKER	OPEN	CLOSE
SWR33	SW PUMP 'D' BREAKER	OPEN	CLOSE
SWR34	SW-6 MN TRAVEL SCREEN WASH	OPEN	CLOSE
SWR35	SW-8 SW HDR TO FIRE MN X-CONN VL	CLOSE	OPEN
SWR36	SW-13A SW SUPLY TO HDR 'A' VLV	CLOSE	OPEN
SWR37	SW-13P SW SUPLY TO HDR 'B' VLV	CLOSE	OPEN
SWR38	SW-18 RX BLG SW DISCH HDR VLV	CLOSE	OPEN
SWR39	RCW-100 DI(MKUP) WTR BYPS LCV-1	CLOSE	OPEN
SWR40	SW-14B SW STRAINER BYPASS VLV	CLOSE	OPEN
SWR41	SW-14E SW STRAINER BYPASS VLV	CLOSE	OPEN
SWR42	RCW-142A AUT FILL ISO VL(RBCCW)	CLOSE	OPEN
SWR43	SEAS SEL-AIR/WTR DEG F	A B	C
SWR44	1C SW PM ALT SHDN XFER SW SS-427	NORM	EMERG
SWR45	1C RHR-SW PM ALT SHDN XFR SW1307	NORM	EMERG
SWR46	1A RHR-SW PM ALT SHDN XFR SW1305	NORM	EMERG
SWR47	1A SW PM ALT SHDN XFR SW SS-425	NORM	EMERG
SWR48	'A' RHR-SW PMP SWITCH (LOC KEYL)	TRIPNORMCLSE	
SWR49	'C' RHR-SW PMP SWITCH (LOC KEYL)	TRIPNORMCLSE	
SWR50	'A' SW PUMP SWITCH (LOC KEYLOCK)	TRIPNORMCLSE	
SWR51	'C' SW PUMP SWITCH (LOC KEYLOCK)	TRIPNORMCLSE	
TCR01	LOCAL MAIN TU MASTER TRIP UNIT	TRIP	RESET
TUR01	AUX OIL PMP LOCAL TEST P.B.	NORM	TEST
TUR02	TURNG GEAR OIL PMP LOCL TEST P.B	NORM	TEST
TUR03	EMERGEN OIL PMP LOCL TEST P.B.	NORM	TEST
TUR04	'A' EPR OIL PUMP	OFF	ON
TUR05	'B' EPR OIL PUMP	HANDOFF	AUTO
TUR07	TEMP FOR TRAINING	FALSE	TRUE

ANALOG IDAS

CDR01	V-73-117 COND PRESS VLV	%	0	100
CDR02	C-145A BYPAS ARND 3A HTR OUTL VL	%	0	100

CDR03	C-145B BYPAS ARND 3B HTR OUTL VL	%	0	100
CDR05	FDW-6A CONDENSATE PMP DISCH VLV	%	0	100
CDR06	FDW-6B CONDENSATE PMP DISCH VLV	%	0	100
CDR07	FDW-6C CONDENSATE PMP DISCH VLV	%	0	100
CDR13	FCV-4 CN RECIRC FLW CNTL VLV	GPM	0	6000
CDR14	FW-30 MANU BYPAS VLV ARND FCV-4	%	0	100
CDR15	FW-18 MKUP/REJ MANU BYPAS VLV	%	0	100
CDR17	AO-SB-8 COND FLTR DEMIN BYPS VLV	%	0	100
CSR01	CS-8A 'A' CS SUCTN FROM CST VLV	%	0	100
CSR02	CS-8B 'B' CS SUCTN FROM CST VLV	%	0	100
CSR03	CS-21A 'A' CS DISCH LN (FLSH)VLV	%	0	100
CSR04	CS-21B 'B' CS DISCH LN (FLSH)VLV	%	0	100
CUR04	CU-FCV-16A 'A' DEMIN FLW CNT VLV	GPM	0	100
CUR05	CU-FCV-16B 'B' DEMIN FLW CNT VLV	GPM	0	100
DGR02	'A' DIESEL GOVRNR DROOP SET		0	100
DGR03	'B' DIESEL GOVRNR DROOP SET		0	100
EGR01	MN GEN H2 PURITY ADJUSTMENT	%	0	100
EGR05	MN GEN H2 MAN PRESS REG ADJUST	PSIG	0	60
FWR01	V-73-117 COND PRESS VLV	%	0	100
FWR02	C-145A BYPAS ARND 3A HTR OUTL VL	%	0	100
FWR03	C-145B BYPAS ARND 3B HTR OUTL VL	%	0	100
FWR04	FDW-37A BYPAS ARND FDW-7A(HP HT)	%	0	100
FWR05	FDW-37B BYPAS ARND FDW-7B(HP HT)	%	0	100
FWR07	FDW-6A CONDENSATE PMP DISCH VLV	%	0	100
FWR08	FDW-6B CONDENSATE PMP DISCH VLV	%	0	100
FWR09	FDW-6C CONDENSATE PMP DISCH VLV	%	0	100
FWR15	FCV-4 CN RECIRC FLW CNTL VLV	GPM	0	6000
FWR16	FW-30 MANU BYPAS VLV ARND FCV-4	%	0	100
FWR17	FW-18 MKUP/REJ MANU BYPAS VLV	%	0	100
FWR19	AO-SB-8 COND FLTR DEMIN BYPS VLV	%	0	100
HPR02	HPCI-23-12 TU EXH MANU ISOL VLV	%	0	100
MCR04	AE-7 COND VAC PMP SUCT ONLY	%	0	100
MCR05	SJAE STEAM SPLY PCV-1 STPT ADJ	PSIG	0	200
NMR61	SRM 'A' GAIN ADJUSTMENT		0.5	1.5
NMR62	SRM 'B' GAIN ADJUSTMENT		0.5	1.5
NMR63	SRM 'C' GAIN ADJUSTMENT		0.5	1.5
NMR64	SRM 'D' GAIN ADJUSTMENT		0.5	1.5
NMR65	IRM 'A' GAIN ADJUSTMENT		0.5	1.5
NMR66	IRM 'B' GAIN ADJUSTMENT		0.5	1.5
NMR67	IRM 'C' GAIN ADJUSTMENT		0.5	1.5
NMR68	IRM 'D' GAIN ADJUSTMENT		0.5	1.5
NMR69	IRM 'E' GAIN ADJUSTMENT		0.5	1.5
NMR70	IRM 'F' GAIN ADJUSTMENT		0.5	1.5
NMR71	APRM 'A' GAIN ADJUSTMENT		0.5	1.5
NMR72	APRM 'B' GAIN ADJUSTMENT		0.5	1.5
NMR73	APRM 'C' GAIN ADJUSTMENT		0.5	1.5
NMR74	APRM 'D' GAIN ADJUSTMENT		0.5	1.5
NMR75	APRM 'E' GAIN ADJUSTMENT		0.5	1.5
NMR76	APRM 'F' GAIN ADJUSTMENT		0.5	1.5
NMR77	RBM 'A' GAIN ADJUSTMENT		0.5	1.5
NMR78	RBM 'B' GAIN ADJUSTMENT		0.5	1.5
OGR01	OG9049 OG-100 LINE DRAIN VLV	%	0	100
OGR06	OG-636 AIR PURGE SUPPLY TO 24IN DLY PIP	%	0	100
OGR07	OG-635A AIR PGE SPLY RECOMB TRN	%	0	100
OGR08	OG-635A AIR PGE SPLY RECMB TRN A	%	0	100
PPR01	INSTRUMENT NOISE ADJUST	%	0	5

RCR02	RCIC-SSC-9 TU EXH MAN ISO VLV	%	0	100
RDR02	CRD-56 CHRGNG WTR HDR STOP VLV	%	0	100
RDR03	CRD-150A 'A' CRD PMP DISCH VLV	%	0	100
RDR04	CRD-150B 'B' CRD PMP DISCH VLV	%	0	100
RDR05	CRD 40/40A CRD PMP TST BYPS VLV	%	0	100
RDR11	'A'RX RECIRC CRD SEAL PRGE REG FL RT	GPM	0	5
RDR12	'B'RX RECIRC CRD SEAL PRGE REG FL RT	GPM	0	5
RHR01	RHR-30A RHR LOOP A PRSRIZNG VLV	%	0	100
RHR02	RHR-30B RHR LOOP B PRSRIZNG VLV	%	0	100
RHR05	RHR-23A 'A'RHR-HX MAN INLET VLV	%	0	100
RHR06	RHR-23B 'B'RHR-HX MAN INLET VLV	%	0	100
RHR21	RHR-28A 'A'RHR-HX MAN OUTLT VLV	%	0	100
RHR22	RHR-28B 'B'RHR-HX MAN OUTLT VLV	%	0	100
RHR23	RHR-2 RHR LOOP A TO B X-TIE VLV	%	0	100
RHR24	RHR70A/71A CN FL VL TO A RHRLPCI	%	0	100
RHR25	RHR70B/71B CN FL VL TO B RHRLPCI	%	0	100
RHR26	RHR-45 CN FL VL TO RHR HD SP LN	%	0	100
RHR27	RHR73/74 CN FL VL TO RC LPRHR SU	%	0	100
RHR28	RHR68 CN FL VL TO RHR56&OR RHR75	%	0	100
RHR29	RHR-56 CN FL VL TO 'B' RHR LOOP	%	0	100
RHR30	RHR-75 CN FL VL TO 'A' RHR LOOP	%	0	100
RHR31	RHR21A/21C RHR PM A&C DISC RW VL	%	0	100
RHR32	RHR21B/21D RHR PM B&D DISC RW VL	%	0	100
RHR33	RHR22A/22C RHR PM A+C SUCT LINE	%	0	100
RHR34	RHR22B/22D RHR PM B+D SUCT LINE	%	0	100
RHR35	RHR11A RHR PM A&C DR TO RADWAST	%	0	100
RHR36	RHR11B RHR PM B&D DR TO RADWAST	%	0	100
RHR42	RHR-17 SHTDN CLG SVLT ISO VLV ALT	%	0	100
RRR07	'A' RECIRC LOOP DISCH TEMP(ADJU)	DEG F	125	540
RRR08	'B' RECIRC LOOP DISCH TEMP(ADJU)	DEG F	125	540
SWR08	SB-1 SW DISCH TO CND DISCH BLK	%	0	100
SWR13	SW-2A SW PMP 'A' DISCHARGE VLV	%	0	100
SWR14	SW-2B SW PMP 'B' DISCHARGE VLV	%	0	100
SWR15	SW-2C SW PMP 'C' DISCHARGE VLV	%	0	100
SWR16	SW-2D SW PMP 'D' DISCHARGE VLV	%	0	100
SWR52	A RECIRC M-G SET L O CLR VLV,V70-22C	%	0	100
SWR53	B RECIRC M-G SET L O CLR VLV,V70-22C	%	0	100
SWR54	RBCCW ISOLATION VALVE 70-118	%	0	100
TUR06	MAIN TU LUBE OIL TANK LVL ADJUST	GALS	0	10000

The controlled copy versions of the plant operating procedures are utilized in the simulator. They are used by instructors in preparing lesson plans and developing scenarios. The students are required to use them during training sessions. The procedures include normal operating, off-normal transients, and emergency plan procedures.

The simulator procedures are controlled and updated to current status through procedural controls similar to those of the reference plant.

DESCRIPTION

This document describes the components making up the simulator design database. It supports the ANSI A2 requirements for simulator certification.

1. The Simulator Design database is made up of the following data bases: Simulator Construction database, Reference Plant database, Simulator As-Built database and the Discrepancy Report (DR) database.

2. The Simulator Construction database consists of the original reference plant documents that were sent to the Simulator manufacturer, Singer Link, for construction of the Simulator.

   2.1 Plant construction and current data available as established per the July 23, 1983, letter of intent were used in both the hardware and software design of the Simulator.

   2.2 Singer Data Requests are also included in this database.

3. The Reference Plant database consists of any Quality Assurance (QA) records generated at the plant including current and historical records.

   3.1 Control of the reference plant records is established by plant procedure AP 6805.

   3.2 Historical records can be obtained through the Plant Document Control Department.

4. The Simulator As-Built database is the information document provided by the Simulator manufacturer documenting the design and construction of the Simulator. It consists of the following:

   4.1 Final Design Specifications.

   4.2 Simulator drawings.

   4.3 Vendor manuals.

   4.4 Simulator Acceptance Test Procedure (ATP).

5. The DR database consists of the changes to the original Simulator Construction database generated by plant modifications or Simulator discrepancies.

   5.1 The DR database identifies, tracks, resolves, and tests differences between the Simulator and the Reference Plant database. All DRs generated from plant modifications should be supported by current reference plant data.

   5.2 The DR database identifies Simulator and computer enhancements. A DR should be written for any change to Simulator hardware or software.

ABSTRACTS

The simulator performance tests are broken down into several major categories:

- Computer real time test
- Steady state and normal operations
- Surveillance tests
- Transient tests
- Malfunction tests

TEST OPERATORS AND DATA REVIEWERS

Name	Position	Qualification
R. Spinney	Training Manager	EdD Education Plant SRO Certified
A. Chesley	Simulator Supervisor	Prior SRO Licensed Presently SRO Certified
G. LeClair	Ops. Training Supv.	SRO Licensed
L. Doane	Asst. Operation Supv.	SRO Licensed
J. Hudachek	Sr. Simulator Analyst	B. Mathematics
M. Krider	Sim. Systems Specialist	BS Computer Science RO Licensed/SRO Certified
D. Tuttle	Sim. Systems Specialist	Prior SRO Licensed Presently SRO Certified
A. Thomas	Simulator Analyst	BA Accounting
S. Brown	Sr. Ops. Instructor	SRO Certified

The base line data used in evaluating the simulator response is specified in the test abstract. The data sources include the Final Safety Analysis Report (FSAR), actual plant data, License Event Reports (LER's), and best estimate judgement. Also, the simulator benchmark test comparing the simulator response to RELAP5YA and RETRAN-02 was used. This test was performed independently by Yankee Atomic Electric Company.

The malfunction test results are compared to the Malfunction Cause and Effects Document. This document is generated from actual plant data and is used by the instructors to develop simulator training scenarios.

The start-up, shutdown, and steady state test requires using the plant operating procedures to maneuver the simulator. After the desired power level is reached, the actual control room round sheets are used to compare the various parameters with the simulator.

VERMONT YANKEE SIMULATOR CERTIFICATION
COMPUTER REAL TIME TEST ABSTRACT

TITLE: Computer Real Time Test (ANSI/ANS-3.5-1985 Section A3.1)

DATE TESTED: October 12, 1990

INITIAL CONDITIONS:

Simulator reset to IC-9 (BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon) with spare time test program TSPARE program running.

ACCEPTANCE CRITERIA:

The Simulator computer divides each second into twenty frames and runs each real time model in one or more of these frames each second. The Simulator computer should have at least 15% average spare time in each frame. No slipped frames (the computer did not have time to complete its work in that frame) should be detected.

PROCEDURE:

Insert malfunctions RR01A (Recirc Loop A Rupture) and ED17 (Loss of Offsite Power) at the same time. Determine the average, as well as the worst case, spare time in each frame for five minutes. Print the results on the line printer.

RESULTS:

All acceptance criteria from above were met. No discrepancy reports were written.

EXCEPTIONS: None

PERFORMED BY: John Hudachek

TITLE: STEADY STATE & NORM. OPS [ANSI/ANS - 3.5 - 1985 B1.1]

DATE TESTED: 12/12/90

INITIAL CONDITIONS:

IC-01 Reactor in cold shutdown condition

PROCEDURE:

A. Reactor start-up to criticality using VYOP 0100. Complete form VYOPF 0100.01, .02, .04.

B. Reactor and generator heat-up to low power using VYOP 0101. Complete forms VYOPF 0101.01, .02.

C. Power operations using VYOP 0102. Stop at power levels of 50%, 75%, 100% and record data specified in the Simulator Critical Parameters Data Sheet.

Reset to IC-9, record data specified in the Simulator Steady State Data Sheets and complete VYAPF 0150.03. Allow the simulator to remain at 100% for 60 minutes. Again record the above specified data.

Using VYOP 0102 decrease reactor power to 20%. Complete VYOPF 0102.01, .02.

D. Shutdown to low power standby using VYOP 0110 and complete VYOPF 0110.01.

E. Shutdown to cold shutdown conditions using VYOP 0111 and complete VYOPF 0111.01.

F. Reset simulator to IC-09 and insert a manual reactor scram. Carry out actions in OE 3100 and VYOP 0109 plant restorations.

G. Perform surveillance testing on safety-related systems using plant specific procedures.

DATA REFERENCE:

Actual plant data, operating procedures, emergency procedures and forms are used for comparison.

RESULTS:

Meets ANSI 3.5 App A3.2 criteria.

See attachment 1 for discrepancies

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

ATTACHMENT # 1

Discrepancy Reports Generated
During Certification Testing
Status as of 12/20/90

INTEGRATED STARTUP, SHUTDOWN AND STEADY STATE TEST

2% Discrepancy Reports Generated

90-0240 AT LOW CORE FLOWS RECIRC PUMP POWER DOES NOT MEET PLANT DATA WITHIN 2%
STATUS - Completed 12/11/90 - Awaiting Retest

10% Discrepancy Reports Generated

90-0229 STATOR COOLING INLET TEMPERATURE
STATUS - Completed and Retested 12/19/90

90-0230 BEARING OIL PRESS
STATUS - To Resolved by the end of the first Qtr 1991

90-0232 CW PUMP "A" AMPERAGE
CW PUMP "B" AMPERAGE
CW PUMP "C" AMPERAGE
STATUS - To Resolved by the end of the first Qtr 1991

89-0146 CORE PRESSURE DROP
STATUS - To Resolved by the end of the first Qtr 1991

90-0234 "A" RECIRC PUMP DIFF PRESSURE
"B" RECIRC PUMP DIFF PRESSURE
STATUS - To Resolved by the end of the first Qtr 1991

90-0235 STACK FLOW
STATUS - To Resolved by the end of the first Qtr 1991

90-0237 RADWASTE EFFLUENT
STATUS - Completed and Retested 12/18/90

90-0239 RX BLDG DIFF PRESSURE "A"
RX BLDG DIFF PRESSURE "B"
STATUS - To Resolved by the end of the first Qtr 1991

ATTACHMENT # 2

Discrepancy Reports Generated
During Certification Testing
Status as of 12/20/90

SAFETY RELATED SYSTEMS SURVEILLANCE TESTING DR's

90-0245 VALVE NG-12A FAILED TO OPERATE CORRECTLY
STATUS - Completed and Retested 12/17/90

90-0258 DIESEL GENERATOR A & B START TIMES EXCESSIVE
STATUS - Completed and Retested 12/19/90

90-0259 FOLLOWING VALVE STROKE TIMES GREATER THAN IST ALLOWABLE:
NG-11A,12A,13A,22A
RHR-13A
CS-7A
HPCI-53
SGT-1A&B,2A&B,3A&B,4A&B
STATUS - Completed 12/19/90 - Awaiting Retest

90-0260 SBGT FAN FLOW EXCESSIVELY HIGH WHEN FLOW CORRECTION FACTORS USED.
STATUS - Completed and Retested 12/4/90

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: MANUAL SCRAM [ANSI/ANS - 3.5 - 1985 App. B1.2(1)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. a full manual scram is inserted through I/O override. The parameters recorded are as per App. B1.2.1. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement and benchmark testing with RELAP5YA and RETRAN-02 are used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. When available actual plant data is utilized.

RESULTS:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: SIMULTANEOUS TRIP OF ALL FEEDWATER PUMPS [ANSI/ANS - 3.5 - 1985 App. B1.2(2)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions FW08A, FW08B, FW08C, (feedwater pumps A,B,C, trips) are simultaneous activated. The parameters recorded are as per App. B1.2.1. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. When available actual plant data is utilized.

RESULTS:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: SIMULTANEOUS CLOSURE OF ALL MAIN STEAM ISOLATION VALVES [ANSI/ANS - 3.5 - 1985 App. B1.2(3)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions RP03 (spurious Group 1 isolation) is activated. The parameters recorded are as per App. B1.2.1. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. Benchmark testing results using RETRAN-02 and RELAP5YA are used in the comparison

RESULTS:

The five minute run time is sufficient for parameter to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: SIMULTANEOUS TRIP OF ALL RECIRCULATION PUMPS [ANSI/ANS - 3.5 - 1985 App. B1.2(4)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions RR05A, RR05B (recirc pump drive motor breaker trips) are simultaneous activated. The parameters recorded are as per App. B1.2.2. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. Benchmark testing results using RELAP5YA and RETRAN-02 along with actual plant data are used in the comparison.

RESULTS:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: SINGLE RECIRCULATION PUMP TRIP [ANSI/ANS - 3.5 - 1985 App. B1.2(5)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions RR05A (recirc pump drive motor A trip) is activated. The parameters recorded are as per App. B1.2.2. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. When available actual plant data is utilized.

RESULTS:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

__TITLE__: MAIN TURBINE TRIP (maximum power level not resulting in a reactor scram) [ANSI/ANS - 3.5 - 1985 App. B1.2(6)]

__DATE TESTED__: 05/30/90

__INITIAL CONDITIONS__:

Simulator reset to IC-7: BOC, 34% Rx Power, 38% Core Flow. Manually reduce Rx power via recirc flow and control rod insertion until the "Stop Valve/Control Valve Bypass" alarm is activated. (the alarm indicates that a turbine trip will not cause a Rx scram)

__PROCEDURE__:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions TC01 (turbine trip) is activated. The parameters recorded are as per App B1.2.1. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

__DATA REFERENCE__:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff. When available actual plant data is used.

__RESULTS__:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

__EXCEPTIONS__: None

__PERFORMED BY__: Dave Tuttle, Mark Krider

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: MAXIMUM RATE POWER RAMP [ANSI/ANS - 3.5 - 1985 App. B1.2(7)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min., using the master controller, manually ramp power down to 75% and back up to 100% hrough CAEF. The parameters recorded are as per App. B1.2.1. The simulator is allowed to run for 5 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff.

RESULTS:

The five minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Dave Tuttle, Mark Krider

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

<u>TITLE</u>: REACTOR COOLANT SYSTEM RUPTURE (DBA LOCA) WITH LOSS OF OFFSITE POWER [ANSI/ANS - 3.5 - 1985 App. B1.2(8)]

<u>DATE TESTED</u>: 05/24/90

<u>INITIAL CONDITIONS</u>:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

<u>PROCEDURE</u>:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions RR01A (recirc loop A rupture 100% severity) and ED17 (loss of offsite power) are simultaneous activated. The parameters recorded are as per App. B1.2.3. The simulator is allowed to run for 10 min. after which time it is placed in "freeze".

<u>DATA REFERENCE</u>:

Benchmark analysis using RELAP5YA, RETRAN-02, and SIMULATE-02 computer codes are used. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff.

<u>RESULTS</u>:

The ten minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

<u>EXCEPTIONS</u>: None

<u>PERFORMED BY</u>: Dave Tuttle, Mark Krider

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: MAXIMUM UNISOLABLE MAIN STEAM LINE RUPTURE [ANSI/ANS - 3.5 - 1985 App. B1.2(9)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunction MS06 (main steam line rupture in drywell) is activated. The parameters recorded are as per App. B1.2.3. The simulator is allowed to run for 10 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, oper·`or training staff, and plant operation department staff.

RESULTS:

The ten minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Dave Tuttle, Mark Krider

VERMONT YANKEE SIMULATOR CERTIFICATION
TRANSIENT PERFORMANCE TEST ABSTRACT

TITLE: SIMULTANEOUS CLOSURE OF ALL MAIN STEAM ISOLATION VALVES WITH STUCK OPEN SAFETY RELIEF VALVE [ANSI/ANS - 3.5 - 1985 App. B1.2(10)]

DATE TESTED: 05/24/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. After 1 min. malfunctions RP03 (spurious Group 1 isolation), AD02A (relief valve "A" stuck open), and HP02 (HPCI failure to start) are activated. The parameters recorded are as per App. B1.2.3. The simulator is allowed to run for 10 min. after which time it is placed in "freeze".

DATA REFERENCE:

Best estimate judgement is used to evaluate the recorded data. The results are subjected to an SRO level review by representatives from the simulator staff, operator training staff, and plant operation department staff.

RESULTS:

The ten minute run time is sufficient for parameters to effectively stabilize. All acceptance criteria (ANSI 4.2.1) were met. No discrepancies found.

EXCEPTIONS: None

PERFORMED BY: Dave Tuttle, Mark Krider

VERMONT YANKEE SIMULATOR CERTIFICATION
MALFUNCTION TEST ABSTRACT

TITTLE: MALFUNCTION TESTING

DATE TESTED: JULY - NOV. 1989

INITIAL CONDITIONS:

Most malfunctions tested from IC-9: BOC, 100% Power, 97% Core Flow, Equilibrium Xenon. Some malfunctions, due to their nature, where tested from lower power levels.

PROCEDURE:

Tests are conducted by monitoring the primary alarms, indicating lights and system parameters as specified in the malfunction Cause an Effects Document.

DATA REFERENCE:

Actual plant data and the Malfunction Cause an Effects Document

RESULTS:

All acceptance criteria (ANSI 4.2.2) were met. The discrepancy reports written during initial testing are identified in Attachment 1.

[See malfunction matrix to cross-reference those malfunction that meet the requirements of ANSI 3.1.2]

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
MALFUNCTION MATRIX

REF: ANSI/ANS - 3.5 - 1985
MALFUNCTION CAUSE & EFFECTS DOCUMENT

The following table of malfunctions is designed to indicate which malfunctions may be used to meet the requirements of section 3.1.2 of the referenced standard.

This table in conjunction with the Vermont Yankee Malfunction Cause and Effects Document will enable the simulator operator to develop the necessary scenarios to meet these requirements.

1. LOSS OF COOLANT

1a. Required for PWRs only.

1b. Inside and outside primary containment

AD01	AD04	CU02	CU03	CU06	FW23
HP09	MS06	MS07	MS08	RC09	RR01

1c. Large and small reactor coolant breaks including demonstration of saturation condition

AD01	AD04	CU02	CU03	CU06	FW23
HP09	MS06	MS07	MS08	RC09	RR01

1d. Failure of safety and relief valves

AD01	AD02	AD04			

2. LOSS OF INSTRUMENT AIR

Loss of instrument air to the extent that the whole system or individual headers can lose pressure and affect the plant's static or dynamic performance

IA01	IA02	IA03			

3. LOSS OR DEGRADED ELECTRICAL POWER

Loss or degraded electrical power to the station, including loss of emergency generators, loss of power to the plant's electrical distribution buses and loss of power to the individual instrumentation buses (AC as well as DC) that provide power to control room indication or plant control functions affecting the plant response

DG01	DG03	DG04	DG05	DG06	ED01
ED02	ED03	ED04	ED05	ED06	ED07
ED08	ED09	ED10	ED11	ED12	ED15
ED16	ED17				

4. LOSS OF FORCED CORE COOLANT FLOW

Loss of forced core coolant flow due to single or multiple pump failure

RR03	RR05	RR06	RR09		

5. LOSS OF CONDENSER VACUUM

Loss of condenser vacuum including loss of condenser level control

CD11	CD13	MC01	MC03	MC04	MC08
MC10	MC11	MC12	MC13	MC14	MC15

6. LOSS OF SERVICE WATER

Loss of service water or cooling to individual components

SW01	SW02	SW03	SW04	SW05	SW06
SW07	SW08	SW09	SW12	SW13	SW14

7. LOSS OF SHUTDOWN COOLING

RH01	RH04	RH05	SW07	SW08	

8. LOSS OF COMPONENT COOLING

Loss of component cooling system or cooling to individual components

SW01	SW14	SW15			

9. LOSS OF NORMAL FEEDWATER

Loss of normal feedwater or normal feedwater system failure

CD01	CD07	CD11	CD13	FW08	FW10
FW11	FW13	FW21	FW23	FW27	

10. LOSS OF ALL FEEDWATER

Loss of all feedwater (normal and emergency)

CD01	CS01	CS02	CS03	FW08	HP01
HP02	HP05	HP07	RC01	RC02	RC05
RD01	RH01	RH06	RH07	SL01	SL02

11. LOSS OF PROTECTIVE SYSTEM CHANNEL

RP02	RP07	RP08	RP09		

12. CONTROL FAILURE

Control rod failure including stuck rods, uncoupled rods, drifting rods, rod drops and misaligned rods

RD02	RD03	RD04	RD05	RD06	RD12

13. INABILITY TO DRIVE RODS

RD01	RD02	RD11	RD15	RD17	RD19

14. FUEL CLADDING FAILURE

Fuel cladding failure resulting in high activity in reactor coolant or off gas and the associated high radiation alarms

RX01					

15. TURBINE TRIP

TC01	TU03				

16. GENERATOR TRIP

EG01	EG04				

17. FAILURE OF AUTOMATIC CONTROL SYSTEMS

Failure of automatic control system(s) that affect reactivity and core heat removal

FW11	RR10	RR11	TC05	TC06	

18. PWRS ONLY

19. REACTOR TRIP

AD04	CD01	CD13	ED16	EG01	EG04
FW08	IA01	IA02	MC01	MC08	MC11
MC15	MS05	MS06	MS07	MS08	MS11
NM03	NM04	NM05	NM2	RP02	RP03
RR01	RR09	RR15	TC01		

20. MAIN STEAM/FEED LINE BREAK

Main steam line as well as main feed line break (both inside and outside containment)

FW21	FW23	FW27	MS06	MS07	MS08
MS11					

21. NUCLEAR INSTRUMENT FAILURE(S)

NM01	NM02	NM03	NM04	NM05	NM06
NM08	NM09	NM2	NM3		

22. PROCESS INSTRUMENTATION FAILURE

Process instrumentation, alarms, and control system failures

AN1	AN2	RM01	RM02	RM03	

23. PASSIVE SYSTEM MALFUNCTIONS

Malfunctions in passive systems, such as engineered safety features, emergency feedwater systems

AD03	CS02	CS03	DG03	DG04	DG05
DG06	FW22	HP01	HP02	HP04	HP07
HP08	MS01	MS02	RC01	RC02	RC03
RD02	RH06	RH07	RP01	RP07	RP08
RP09	SL01	SL02			

24. FAILURE OF AUTOMATIC REACTOR TRIP SYSTEM

RP01					

25. REACTOR PRESSURE CONTROL FAILURE

Reactor pressure control system failure including turbine bypass failure

RP01	TC02	TC03	TC05	TC06	

ATTACHMENT # 3

Discrepancy Reports Generated
During Certification Testing
Status as of 12/20/90

MALFUNCTION TESTING

90-0079 MALFUNCTION HP01 - SYSTEM READY LITE PROBLEM
STATUS - Completed and Retested 5/13/90

90-0079 MALFUNCTION RC06 - NO TEMPERATURE RISE
STATUS - To Resolved by the end of the first Qtr 1991

90-0299 NO MALFUNCTION EXISTS TO FAIL THE SAFETY VALVES OPEN
STATUS - Completed 12/18/90 - Awaiting Retest

VERMONT YANKEE SIMULATOR CERTIFICATION
CRITICAL OPERATING LIMIT TEST ABSTRACT

TITLE: SIMULATOR OPERATING LIMITS [ANSI/ANS - 3.5 - 198^ 4.3]

DATE TESTED: 9/28/90

INITIAL CONDITIONS:

Simulator reset to IC-9: BOC, 100% Rx Power, 97% Core Flow, Equilibrium Xenon

PROCEDURE:

Initial conditions are established and the simulator placed in run. Reduce or increase the design parameter limit, one at a time, to just above the present value in IC-9. Ensure the two "SIMULATOR OPERATING LIMIT EXCEEDED" lights are out. Appropriately increase or decrease the present value through simulator manipulations until the "SIMULATOR OPERATING LIMIT EXCEEDED" light comes on.

The design parameter test are:

* Reactor Vessel Internal Pressure
* Primary Containment Internal Pressure
* Primary Containment External Pressure

DATA REFERENCE:

Reference Plant design limits

RESULTS:

All acceptance criteria is satisfactory

EXCEPTIONS: None

PERFORMED BY: Mark Krider, Dave Tuttle

VERMONT YANKEE SIMULATOR CERTIFICATION
PERFORMANCE TESTING SCHEDULE

EACH YEAR THE FOLLOWING TESTS WILL BE PERFORMED:

1. Computer Real Time Test

2. ANSI/ANS 3.5 - 3.1.1/app. A3.2 Steady State and Normal Operation Test

3. ANSI/ANS 3.5 - A3.3 Transient Tests:

   a. ANSI B1.2(1) Manual Scram

   b. ANSI B1.2(2) Simultaneous Trip Of All Feedwater Pumps

   c. ANSI B1.2(3) Simultaneous Closure Of All Main Steam Isolation Valves

   d. ANSI B1.2(4) Simultaneous Trip Of All Recirc Pumps

   e. ANSI B1.2(5) Single Recirculation Pump Trip

   f. ANSI B1.2(6) Main Turbine Trip

   g. ANSI B1.2(7) Maximum Rate Power Ramp

   h. ANSI B1.2(8) Reactor Coolant System Rupture

   i. ANSI B1.2(9) Maximum Unisolable Main Steam Line Rupture

   j. ANSI B1.2(10) Simultaneous Closure Of All Main Steam Isolation Valves with Stuck Open Safety Relief Valve

4. ANSI/ANS 3.5 - A3.4 Malfunction Tests

   a. All the simulator malfunctions will be tested in a four year cycle. 1991 will be the first year following the initial certification. Attachment 3 identifies which malfunctions are scheduled in a given year.

b. Starting in 1991, 25% of all malfunctions will be tested annually in accordance with this schedule.

1991		1992	
ANNUNCIATORS (AN)	2	SLC (SL)	3
CORE SPRAY (CS)	4	RWCU (CU)	6
TURBINE (TU)	3	INSTRUMENT AIR (IA)	5
PRIMARY CONT. (PC)	10	RHR (RH)	7
MAIN STEAM (MS)	12	TURBINE CONTROL (TC)	9
NUCLEAR MONITORS (NM)	14	ELECTRICAL DIST (ED)	17
CONTROL RODS (RD)	19	REACTOR RECIRC (RR)	17
Total Malf.	64		64

1993		1994	
AUTO DEP. SYS. (AD)	6	RX MAN. CONTROL (RM)	3
DIESEL GEN. (DG)	7	ROD WORTH MIN. (RW)	2
RX PROTECTION (RP)	9	ROD WORTH (RX)	2
MAIN GENERATOR (EG)	10	HPCI (HP)	9
MAIN CONDENSER (MC)	16	RCIC (RC)	9
SERVICE WATER (SW)	16	OFF GAS (OG)	10
		FEEDWATER (FW)	15
		CONDENSATE (CD)	12
Total Malf.	64		62

VERMONT YANKEE SIMULATOR CERTIFICATION
SIMULATOR CONFIGURATION MANAGEMENT SYSTEM

The simulator configuration management system (SCM), in part, is used to generate, track, and store discrepancy reports (DR's). A DR can be generated by anyone including instructors, students, or the simulator support group. The person generating the DR can do so using a DR form or by direct entry into the computer through the SCM system.

The purpose of DR's is to note deficiencies in physical and functional fidelity and track them through resolution, testing, and closeout. DR's are also used to track engineering design changes (EDCR's) and plant design changes (PDCR's) which affect the simulator. The following steps are taken for all DR's:

1. Originator generates the DR.
2. Senior Simulator Analyst (SSA) assigns a Cognizant Engineer (CE) and a Fidelity Assessor (FA).
3. The CE decides if the DR is valid. If the DR is considered not valid the FA must also agree and the SSA will close it out.
4. The CE will perform the work needed to resolve valid DR's.
5. The FA will perform and document an acceptance test.
6. Upon satisfactory completion of testing, the SSA will ensure all documentation is updated and close out the DR.

The SCM is a computer-based program which, in addition to tracking DR's, tracks plant setpoint changes and supports the simulator usage schedule. The SCM program is controlled through procedures contained in the Simulator Administration Manual.

MEMORANDUM

YANKEE ATOMIC - FRAMINGHAM

To: D. A. Reid
Date: July 5, 1988
From: M. J. Marian
Group #: OPVY 503/88
W.O. #: 4100 (PM No. 368A)
Subject: SIMULATOR BENCHMARKING ANALYSIS
I.M.S. #:

Attachments:

1. Benchmark Analysis of the Vermont Yankee Control Room Simulator Using YAEC Best Estimate Analysis, dated June 1988.

Discussion:

Attached, please find an unbound copy of the final simulator benchmarking report. Bound copies are currently being prepared and will be forwarded shortly.

M. J. Marian

M. J. Marian
Vermont Yankee Project Engineer

MJM/sv

cc: SRMiller - w/o
TPWhite
KEStJohn - w/o
MJCofske - w/o

RECEIVED
JUL 7 1988
OPS. SUPPORT

BENCHMARK ANALYSIS OF THE VERMONT YANKEE CONTROL ROOM SIMULATOR USING YAEC BEST ESTIMATE MODELS

By

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R. Thomas Fernandez
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June 1988

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Approved By: B C Slifer 6/30/88

B. C. Slifer, Director (Date)
Nuclear Engineering Department

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ABSTRACT

This report presents the results of the benchmarks performed on the Vermont Yankee Control Room Simulator in May 1987. The benchmarks consisted of a series of operational maneuvers, transients, and accident scenarios which were previously simulated by use of best estimate engineering tools at the Yankee Nuclear Services Division. A comparison of the Simulator's predicted response and the engineering codes' predicted response is provided.

ACKNOWLEDGMENTS

The authors would like to acknowledge the participation of the Vermont Yankee Simulator training and support staff and other members of the YNSD NED staff in the preparation of the benchmarks. The Simulator staff under A. R. Chesley included D. E. Tuttle, A. F. Thomas, R. F. Slauenwhite, D. L. Moran, M. R. Krider, and J. T. Hodachek. Their help in preparing the Simulator, performing the benchmark scenarios, and discussing the results proved invaluable to our understanding of the nuances of the Simulator's predicted response. The NED staff supporting the benchmark effort from the initial benchmark testing in 1984 through the final benchmark included K. J. Burns, J. N. Loomis, M. P. LeFrancois, and J. Pappas.

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1.0 Introduction

The Vermont Yankee Nuclear Power Corporation (VYNPC) contracted with the Link Simulation Division of the Singer Co. for the construction of a full scope control room simulator for the Vermont Yankee Nuclear Power Station (VYNPS) in 1983. The VYNPS is a 1593 MWth BWR/4 with a Mark I containment. The Simulator was delivered to the VYNPC training center in 1986. The Simulator benchmark effort documented in this report supplements the Acceptance Test Program. The Simulator is required to undergo a certification process before it can be employed in NRC administered operator examinations. The benchmarks documented in this report are part of the certification process.

1.1 Simulator Description

The VYNPS simulator is a complete replica of the VYNPS control room. The Simulator provides fully operable front panel instrumentation and controls. Additionally, the Simulator provides at least one operable channel of all instrumentation and controls on the back panels in the control room. The non-operable back panels, however, retain full visual replication. The Simulator also includes an operable replica of the remote RCIC/RHR control panel and one of the remote diesel generator control panels. The Simulator updates all instrumentation displays and monitors all control switches on a basic cycle time of 0.25 seconds.

The Simulator complex also includes two GOULD 32/8750 mini-computers. Operation of the Simulator complex requires one of the 32/8750s. The remaining unit is either in standby or in use for software development. Each mini-computer includes a CPSI 3300 vector processor. The two mini-computers share four CDC 300Mb disk drives, two Kennedy 91 tape drives, and two Pertec 9571-2 tape drives, as well as the Simulator's instructor console, a number of interactive terminals, and line printers. The GOULD operating system version 3.2 is employed.

1.2 Benchmark Strategy

The Vermont Yankee Control Room Simulator Benchmark effort was part of a larger effort to verify that the Simulator did provide the reactor operating crew a realistic representation of the control room during all phases of reactor operations. The benchmark effort focused on the primary system since this is the area that the operator is expected to be focusing on during operator training for response to off-normal conditions. The thermal-hydraulic benchmark scenarios employed in the benchmark were:

1. Large Break LOCA - The design basis accident for the VY plant.
2. Small Break LOCA - The design basis accident with HPCI as the limiting single failure.
3. Main Steam Line Break - The break was the complete rupture of one of the four steam lines outside of the containment.
4. ATWS - The closure of the Main Steam Line Isolation Valves (MSIV) was followed by failure of the control rod drive system such that shutdown required the Standby Liquid Control System (SLCS) to function.
5. RPT Trip - The transient was initiated by the simultaneous tripping of the drive motor in both recirculation loops.

An additional set of benchmark scenarios were performed to validate the core neutronics module. These benchmarks were:

1. Rod pattern exchange - The Simulator was taken through a complete pattern exchange from an A1-2 to a B1-1 pattern.
2. Partial SCRAM - The Simulator was set to prevent the complete insertion of the control rods such that

power was significantly reduced but the plant was still critical.

Several intermediate benchmark runs were performed as the Simulator was being built by Singer-Link. These early efforts were aimed at providing insight into the modeling techniques employed by Singer-Link.

The benchmark effort consisted of comparing the dynamic Simulator response with precalculated responses from Yankee Nuclear Services Division (YNSD) best estimate engineering calculations. The YNSD calculations employed the RELAP5YA, RETRAN-02, and SIMULATE-2 (References 1-2.1, 1-2.2, 1-2.3) computer codes.

1.3 Engineering Code Overview

The SIMULATE computer code was used to provide initial power distributions and other required data to the thermal-hydraulic systems code. SIMULATE was also used to predict the plant response during the Partial SCRAM event.

The RELAP5YA and RETRAN-02 computer codes were the major engineering codes employed in the analysis of the thermal-hydraulic behavior for the various transients and accidents. Both codes divide the system being modeled into a number of user specified volumes with each volume having its own thermodynamic state. The individual volumes are joined by means of junctions describing the flow paths within the system. The user also specifies a number of boundary conditions for systems which are not modeled. These systems include the feedwater and turbine systems of the secondary plant at the VYNPS.

Appendix A gives a more detailed overview of the engineering models employed in the benchmarking.

1.4 Data Acquisition

The acquisition and the display of the Simulator predicted response was accomplished by means of three computer codes. The three computer codes were RXTAPE, SELREAD, and SELPLOT. Figure 1-4.1 illustrates the relationships of these codes.

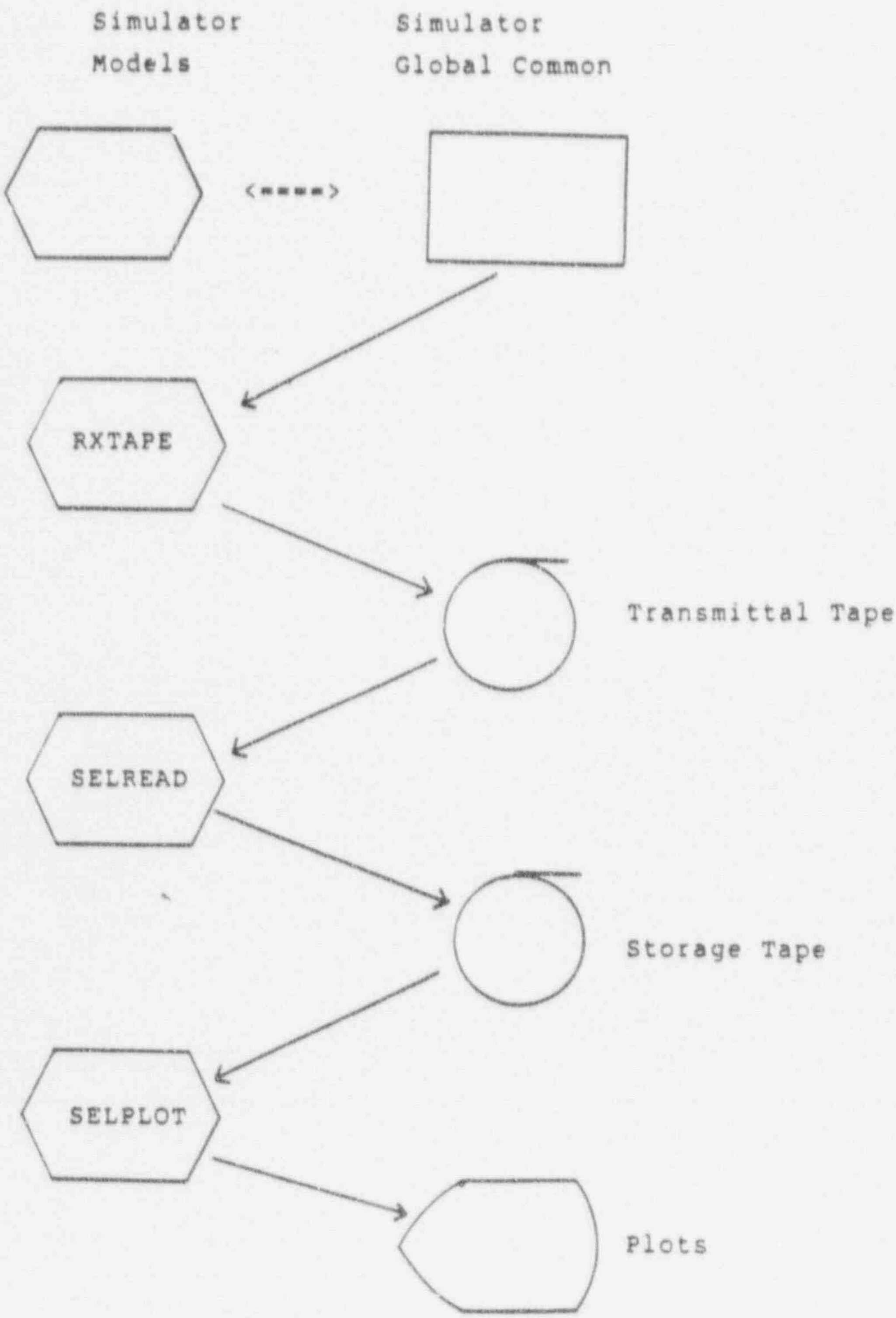
RXTAPE, the first computer code, was run as a scheduled task on the Simulator computer and recorded data from the Simulator Global Common at 0.25 second intervals (Simulator cycle time) onto magnetic tape. The Simulator Global Common is the Simulator's internal data structure for storage of all variables used in the various models which comprise the logic driving the Simulator. These internal variables include variables which are not displayed on the control board. A total of 453 data items were recorded at each interval as a series of binary values.

SELREAD, the second computer code, was run offline at the YNSD CDC computer center on a CDC Cyber-180/855. The function of the code was to read the magnetic tape created by the RXTAPE program and convert the binary data from the GOULD 32/8750 internal format to a format that could be employed by the SELPLOT program and other engineering tools. The SELREAD program also performed data conversion from a stream on N packets containing 453 variables to a stream of 453 packets containing N intervals. The converted data was stored on magnetic tape for future use.

SELPLOT, the final program, was also run offline and plotted the Simulator variables as a function of time. SELPLOT employed the DISSPLA graphics subroutine library to perform the actual plotting.

Figure 1-4.1

COMPUTER CODE INTERACTION



2.0 Reactor Core Physics Model Benchmark

The reactor core model in the Simulator includes a neutronics model which is basically the synthesis of a two dimensional radial flux calculation and a one dimensional axial flux calculation. The two benchmarks chosen for this section were intended to challenge the computational ability of the reactor core physics model.

The SIMULATE-2 code, used extensively at YNSD, served as the engineering code for use in the Partial SCRAM analysis.

2.1 Rod Pattern Exchange

To provide a more uniform burnup of fuel, Vermont Yankee alternates between A1, B1, A2, and B2 sequences throughout a cycle. These sequence exchanges typically are performed every 1000 to 1200 MWD/ST of cycle exposure. The exchanges are performed at approximately 50% of core rated power. The exchange is performed as follows:

1. Power is reduced from full power to approximately 50% of rated power by reducing recirculation flow to its minimum, while maintaining the full power pre-exchange control rod pattern.

2. At minimum flow, the control rods are maneuvered to the full power post-exchange pattern. The actual exchange at the plant follows a specific algorithm to insure that fuel preconditioning limits are not violated when previously controlled fuel segments are uncovered by rods.

3. After the full power post-exchange pattern is obtained, power is increased to full power by increasing recirculation flow.

In evaluating the Simulator performance for the sequence exchange at power, the following conditions should be answered:

1. Does the Simulator accurately model the core reactivity

for the pre-exchange and post-exchange rod patterns. That is, does the core closely approximate the power and flow values at close to full flow?

2 Does the Simulator closely match the change of reactivity versus flow for both the pre-exchange and post-exchange patterns? Does the Simulator predicted power approximate the plant power at minimum flow for the exchange sequence?

The sequence exchange was an actual Cycle 9 A1-2 to B2-2 sequence exchange (Reference 2-1.1). The pre-exchange A1-2 and post exchange B2-2 patterns are shown in Figures 2-1.1 and 2-1.2. The actual exchange took place at a cycle exposure of 4400 MWD/ST. Since this was a period in the cycle of flat reactivity versus exposure, the Simulator middle of cycle conditions at 3800 MWD/ST should accurately model the plant response for this exchange.

The Simulator A1-2 sequence power was reduced by decreasing flow from 47.4 Mlbs/Hr to 18.3 Mlbs/Hr. This flow reduction was performed within a few minutes, whereas the flow reduction took over an hour for the actual plant sequence exchange. The control rod exchange on the Simulator was performed somewhat faster than the plant exchange, because preconditioning constraints were not necessary. Following the exchange, the B2-2 power was increased by increasing reactor core flow. The fast xenon buildup option of the Simulator was initiated during the flow ramp to model the actual xenon buildup during the longer duration of the plant flow ramp.

Table 2-1.1 compares the plant and Simulator response for the exchange. The comparison of the results was very favorable.

2.2 Partial SCRAM results

Currently the Partial SCRAM capability of the Simulator differs from the Partial SCRAM analysis performed for benchmarking. The analysis of the Partial SCRAM attempted to obtain a reactor power of greater than 10 percent power for an event where the control rods in one half of the core insert fully and the control rods in the the other half of the core only inserted to a predetermined maximum length of travel. For example, if the maximum length of travel is specified as 96 inches, rods withdrawn 96 inches or more would insert 96 inches. Rods withdrawn less than 96 inches would insert to their full in position.

The Partial SCRAM was analyzed using a half core SIMULATE model which was modified to include a simplified decay heat model. The decay heat model was required to estimate the resulting water density following the SCRAM. Without this modification, SIMULATE would add only fission heat to the moderator, thereby underpredicting void fraction and overpredicting core reactivity.

The Partial SCRAM was modeled for the all-rods-out condition at end of Cycle 9. To obtain a reactor power level of greater than 10 percent after SCRAM required the partial insertion limit to be 96.0 inches. For the all-rods-out case, this resulted in the control rods in one half of the core inserting fully and rods in the other half of the core inserting to position 16 (Figure 2-2.1). The Simulator currently contains an insertion limit of 114 inches which results in a rod initially fully withdrawn inserting to notch position 10. A Partial SCRAM consisting of half the core inserting fully and half the core limited to a maximum insertion of 114 inches results in the reactor becoming subcritical.

To simulate a Partial SCRAM where the reactor remains at a power level above 10 percent, the Simulator was scrammed with an insertion of 114 inches applied to both halves of the core. The conditions prior to the SCRAM were the A1-2 control rod pattern at

MOC conditions used for the control rod sequence exchange at power. The resulting Partial SCRAM pattern is shown in Figure 2-2.2.

The results of both the analyzed Partial SCRAM and the actual Partial SCRAM modeled on the Simulator are shown in Table 2-2.1. Although the actual Simulator case differs from that analyzed, it shows similar power versus flow behavior and results in a final power above 10 percent.

TABLE 2-1.1

ROD PATTERN EXCHANGE RESULTS

	Plant		Simulator	
	Power (%)	Flow Rate (Mlbs/Hr)	Power (%)	Flow Rate (Mlbs/Hr)
Pre-exchange A1-2 seq.	100.0	46.8	100.0	47.4
	53.0	17.76	51.2	18.34
Post-exchange B2-2 seq.	50.8	17.76	48.9	18.35
	75.0	32.0	77.0	32.0
	84.5	35.0	82.5	35.0
	100.0	45.12	96.0	45.1

TABLE 2-2.1

PARTIAL SCRAM RESULTS

Analyzed case		Simulator	
Power (%)	Flow Rate (Mlbs/Hr)	Power (%)	Flow Rate (Mlbs/Hr)
35.3	52.8	27.0	52.5
14.8	17.8	12.0	15.9

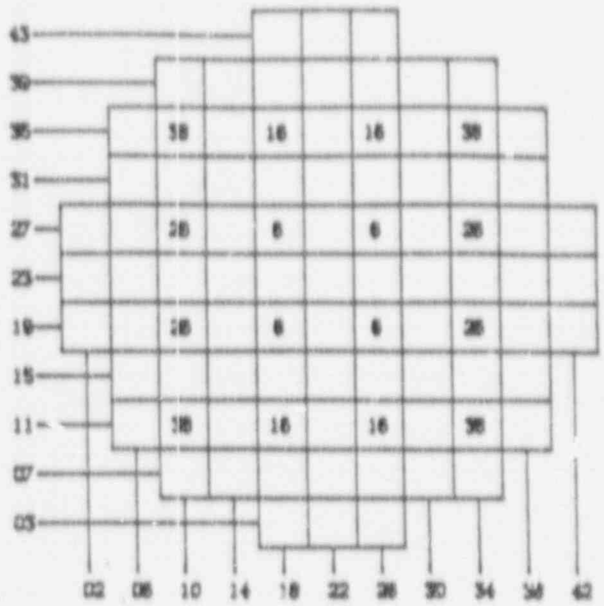


Figure 2-1.1 Pre-exchange Rod Positions

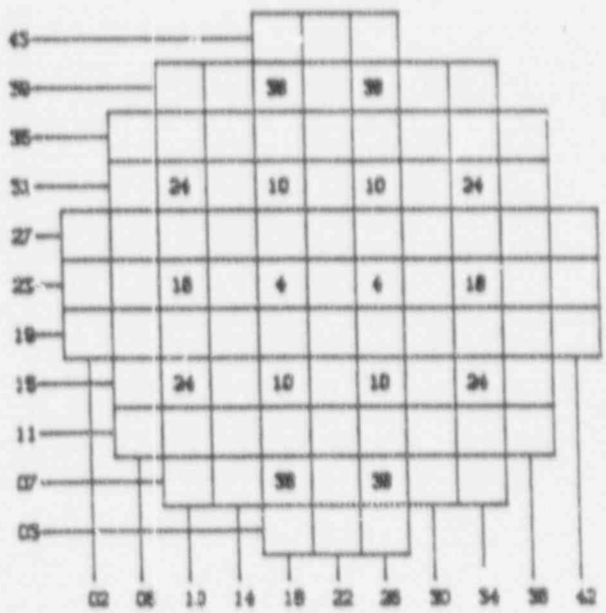


Figure 2-1.2 Post-exchange Rod Positions

	02	06	10	14	18	22	26	30	34	38	42
43					16	16	16				
39			16	16	16	16	16	16	16		
35		16	16	16	16	16	16	16	16	16	
31		16	16	16	16	16	16	16	16	16	
27	16	16	16	16	16	16	16	16	16	16	16
23	0	16	0	16	0	16	0	16	0	16	0
19	0	0	0	0	0	0	0	0	0	0	0
15		0	0	0	0	0	0	0	0	0	
11		0	0	0	0	0	0	0	0	0	
07			0	0	0	0	0	0	0		
03					0	0	0				

Figure 2-2.1 SIMULATE Post-SCRAM Rod Positions

	02	06	10	14	18	22	26	30	34	38	42
43					10	10	10				
39			10	10	10	10	10	10	10		
35		10	0	10	0	10	0	10	0	10	
31		10	10	10	10	10	10	10	10	10	
27	10	10	0	10	0	10	0	10	0	10	10
23	10	10	10	10	10	10	10	10	10	10	10
19	10	10	0	10	0	10	0	10	0	10	10
15		10	10	10	10	10	10	10	10	10	
11		10	0	10	0	10	0	10	0	10	
07			10	10	10	10	10	10	10		
03					10	10	10				

Figure 2-2.2 Simulator Post-SCRAM Rod Positions

3.0 TRANSIENT BENCHMARKS

Two of the scenarios considered for the benchmark were Reactor Recirculation Pump Trip (RPT) and Anticipated Transient Without SCRAM (ATWS). The initial plant conditions and equipment setpoints assumed in these transients are documented in Reference 3-0.1. The RPT simulation was compared to the VY Cycle 8 RPT/Stability tests and the ATWS simulation was compared to best-estimate Vermont Yankee (VY) ATWS RETRAN model predictions, as stated in Reference 3-0.2.

3.1 Reactor Recirculation Pump Trip Benchmark

This section documents the Reactor Recircualtion Pump Trip (RPT) comparison. It consists of a benchmark between the Simulator and the VY Cycle 8 RPT/Stability tests.

3.1.1 VY Cycle 8 RPT/Stability Tests

The VYNPS Cycle 8 RPT/Stability tests were employed since the data was readily available and the use of actual plant data provides a non-biased database.

3.1.2 RPT Event

The event analyzed is the simultaneous manual tripping of both recirculation loop Motor/Generator (M/G) set drive motors from the control board. The transient was initiated from approximately 84% power/flow conditions in order to match the Cycle 8 RPT/Stability test data. All reactor protection systems were assumed to be operational in automatic mode. Other than the initial manual trip of the M/G sets, no operator actions were assumed.

3.1.3 RPT Sequence

The plant was assumed to experience a simultaneous trip of both recirculation loop M/G sets, done manually from the control board approximately 6 seconds from the start of data recording. This causes the recirculation pumps to coastdown, thus placing the plant in a natural circulation mode. With reduced core flow, the core region void fraction increases. This effect produces a significant amount of negative reactivity. As a result of these changes, the core power is rapidly reduced. The entire transient takes less than one minute. During that time the plant stabilizes to approximately 40% power and 30% flow (natural circulation conditions).

3.1.4 RPT Comparison

Table 3-1.1 contains a comparison of the initial operating conditions assumed in the RPT simulation compared to the Cycle 8 RPT/Stability test. Figures 3-1.1 through 3-1.7 depict plant responses calculated by the Simulator overlaid with actual plant data for the RPT transient. These figures show that the Simulator is predicting the correct trends for the RPT simulation, and with fairly good accuracy on most parameters. However, there are a couple parameters, such as core power and M/G speed, where the Simulator could be improved. The remainder of this comparison entails a short discussion of the RPT benchmark parameters.

Figure 3-1.1 displays Core Thermal Power. This is actually core neutron power produced by neutron fission. The recirculation pump trip causes a sharp reduction in core flow. This creates significant voiding, and thus negative reactivity, thereby reducing core power. The Simulator does not decrease as rapidly or as low as the test data. This could be attributed to a variety of reasons; however, the two most outstanding are: 1) lack of a core bypass model; and 2) Inadequate M/G speed coastdown characteristics.

The Simulator models the fluid in the core bypass region and the fuel region as one volume. For Vermont Yankee, the bypass region constitutes approximately 63% of the total core volume. This region is shielded from much of the fission generated heat due to the fuel assembly channels. Therefore, a large percentage of the void change experienced inside the assembly channels (i.e. the fuel region) is not seen in the bypass area. If these distinct regions are combined mathematically as in the Simulator, the predicted void formation and thus reactivity feedback, will be dampened. For the case of a RPT, the associated void formation will insert large amounts of negative reactivity into the core, thus sharply reducing core power. The Simulator does not predict a power reduction as steep as the data for the RPT benchmark due to the lack of a separate bypass volume.

The M/G set speed coastdown characteristics indirectly affect core power due to the consequence of the recirculation pump de-energizing and reducing core flow. The inadequacy of the M/G set speed coastdown characteristics are discussed under Figure 3-1.7, M/G Speed. The Simulator stabilizes to approximately 43% power. This is slightly higher than the plant data but is considered to be acceptable.

Figure 3-1.2 displays Steam Dome Pressure. The Simulator is predicting a slightly slower pressure decrease. Pressure is tightly coupled to core power, and therefore, is remaining high for the same reasons discussed above. However, at no time does the pressure calculated by the Simulator vary more than 8 psi from the plant data. This is considered to be in very good agreement.

Figure 3-1.3 shows Narrow Range Water Level. The Simulator fails to predict the initial 7 inch reactor vessel level swell.

This can be attributed in part to the slightly dampened pressure response observed under Figure 3-1.2. Although this is important from an operator awareness view, it is insignificant for the overall transient and is considered to be acceptable. After the initial swell, the Simulator continuously predicts a slightly higher vessel water level; this may be due to a minor difference in the Three-Element Level Control dynamic compensator. However, at no time is the level difference greater than 2 inches above the plant data. This is considered to be in very good agreement.

Figure 3-1.4 shows Core Inlet and Bypass Flows. The Simulator predicted core flow matches extremely well in comparison to the plant data. The Simulator calculates the flow stabilizing at precisely the same time at approximately 33% flow. This is slightly higher than the plant data but is considered to be acceptable.

Figure 3-1.5 is a plot of Main Steam Line Flow. The Simulator is predicting a slightly slower main steam flow decrease. Steam flow is tightly coupled to core power, and therefore, remains high for the same reasons discussed previously. However, at no time does the flow calculated by the Simulator vary more than 80 lb/sec from the plant data. This is considered to be in very good agreement.

Figure 3-1.6 displays Feedwater Temperature. The initial feedwater temperature predicted by the Simulator is approximately 4 degrees Fahrenheit lower than the data. However, in most transients this minor difference would be insignificant. The important observation is that the Simulator calculates a decreasing temperature to characterize less steam flow being passed through the turbines, and less extraction steam used by the feedwater heaters.

Figure 3-1.7 displays Reactor Recirculation Loop M/G Set Speed. The Simulator does not model the M/G speed coastdown adequately. The predicted coastdown is too slow and ends too low. This directly effects the recirculation flow rate and, consequently, the core flow rate. Since the core flow rate governs much of the void formation and thus the core power reduction, the M/G characteristics indirectly effects the core power predictions.

The Simulator M/G coastdown does not correctly model the field breaker trip that occurs approximately twenty-eight seconds after tripping the M/G sets.

3.2 ATWS Benchmark

This section documents the ATWS comparison. It consists of a benchmark between the Simulator and the VY ATWS RETRAN model.

3.2.1 ATWS RETRAN Model for VY

The best estimate VY RETRAN ATWS Model is discussed in Appendix A.1.

3.2.2 ATWS Event

The event analyzed is an inadvertent closure of all Main Steam Isolation Valves (MSIVs) initiated from 100% power/flow conditions at End Of Full Power Life (EOFPL) with a complete failure of the automatic reactor trip function. No credit was taken for manual control rod insertion. The MSIV closure with no control rod motion constitutes the most severe ATWS event. In addition, no credit was taken for Control Rod Drive (CRD) or Reactor Core Isolation Cooling (RCIC) flow. The High Pressure Coolant Injection (HPCI) and pressure relief valves were maintained in automatic mode. Operator action was assumed for disabling the Automatic Depressurization System (ADS) and initiating the Standby Liquid Control System (SLCS) two minutes after MSIV closure. A thirty second delay was assumed before the sodium pentaborate solution reached the reactor vessel core.

3.2.3 ATWS Sequence

At approximately sixteen seconds from start of Simulator data recording, the plant was assumed to experience a simultaneous closure of all MSIVs. This causes the reactor vessel pressure to rapidly increase and, consequently, creates the collapse of some voids. Void collapse inserts positive reactivity thus increasing reactor power. That in turn causes increased steam generation and further increases pressure. The positive feedback effect continues until the steam line pressure increases to the pressure relief valve setpoints. The valves open, the reactor vessel pressure increase is terminated and pressure is reduced. The recirculation pumps are automatically tripped on high pressure, and core flow is reduced to between 20% and 30% of rated conditions.

With reduced flow, the moderator temperature at the core inlet region is increased. This effect combined with the lower reactor vessel pressure produces voids and large amounts of negative reactivity. Thus the core power rapidly decreases to approximately 30% of rated conditions. the initial power spike also causes a large swell in reactor vessel water level, which trips the feedwater pumps on a high water level signal.

The ECCS pumps auto-start on high drywell pressure. HPCI is the only pump capable of injecting against the RPV pressure and it has a normal startup time of approximately twenty-five seconds before full HPCI flow is achieved. The HPCI turbine stop valves close on receipt of a high reactor water level signal. Once the stop valves are closed, HPCI remains off until the trip signal is cleared by a low water level signal. Depending on the rate of the initial level swell, HPCI might not inject any water since a trip on high reactor water level may be received almost simultaneously.

3.2.4 ATWS Comparison

Table 3-2.1 contains the initial operating conditions and Table 3-2.2 lists the trips and operator actions assumed in the ATWS simulation for the RETRAN and Simulator transients. Figures 3-2.1 through 3-2.8 depict plant response calculated by the RETRAN code and the Simulator. These figures show that the codes are predicting the same plant trends up to 180 seconds after the MSIV closure. After 180 seconds there are two major differences: 1) RETRAN predicts a delayed increase in downcomer water level where the Simulator calculates an immediate increase in water level (both codes predict that HPCI is at full flow by 180 seconds); a delayed level increase is reasonable since HPCI is initiated on a low water level signal, thereby, injecting subcooled water into the steam region that induces steam condensation; and 2) the Simulator predicts that recirculation coastdown trends to zero flow where RETRAN predicts a quasi constant recirculation flow of approximately 400 lb/sec per loop. Continued flow in the recirculation lines seems reasonable since the plant is designed to bypass flow around the idle recirculation pumps in order to eliminate the possibility of large moderator feedback from an inadvertent pump restart. The remainder of this comparison entails a short discussion of the ATWS benchmark parameters.

Figure 3-2.1 shows Core Thermal Power. This is actually core neutron power produced by neutron fissions. Both codes predict the initial power spike due to sudden closure of the MSIVs, and a decrease in power to approximately 30% of rated conditions following the recirculation pump trip. The irregular behavior of the power is due to void reactivity changes caused by pressure fluctuations. The RETRAN predicted core power is more irregular than the Simulator calculated power. This is attributed to the more sophisticated void reactivity feedback model in RETRAN. However, the integrated power is a more important parameter, since it determines the total amount of energy produced in the core which is deposited in the Pressure Suppression Pool (PSP). Although the integrated power was not explicitly calculated, the power parameter looks reasonable upon visual inspection. Decay heat levels are achieved by injecting the required amount of sodium pentaborate solution enabling the plant to reach hot shutdown. The RETRAN calculated power decreases to decay heat levels after 1050 seconds where the Simulator predicts decay heat levels to occur after 1300 seconds. This is attributed to the difference in the SLCS injection capacities assumed for the two codes. The second power swell observed in the Simulator is attributed to the second major actuation of the HPCI system.

Figure 3-2.2 shows Reactor Vessel Steam Dome Pressure. Both codes predict the initial pressure spike due to the closure of the MSIVs. The behavior of the pressure and core power parameters are tightly coupled; thus the pressure spike corresponds to the power spike. The pressure (and power) reduces after the pressure relief valve setpoints are reached and the dome pressure decreases to approximately 1095 psia (the lowest relief valve setpoint). The response of the Simulator is in good agreement with RETRAN. The larger pressure dips predicted by the Simulator can be attributed to the lower relief valve closure setpoints modeled in the Simulator as compared to those modeled in RETRAN.

Figure 3-2.3 displays Reactor Vessel Wide Range Water Level. Both codes predict an initial level swell due to a core power spike, followed by a decrease in vessel inventory caused by the feedwater pump trip in conjunction with continual mass removal through the relief valves. RETRAN predicts a delayed increase in water level from HPCI where the Simulator calculates an immediate increase. A delayed increase in vessel water level is reasonable since HPCI is restarted on low water level. HPCI is injected through the feedwater sparger which is located three feet above the low level initiation setpoint. When the water level falls below the sparger, the HPCI system injects subcooled water into the steam region. This induces sudden steam condensation resulting in the depressurization of the vessel. Typically, water droplets will reach saturation after a fall of about two feet below the feedwater sparger. The Simulator is not capable of can predicting this phenomena because the model injects HPCI directly into the core.

Once decay heat levels are reached, the reactor vessel water level stabilizes. This occurs after 1050 seconds in RETRAN and after 1300 seconds in the Simulator, and corresponds to the SLCS injection capacities assumed for the two codes. The second level swell observed in the Simulator is attributed to the second actuation of the HPCI system.

Figure 3-2.4 displays Core Inlet and Bypass Flow. The initial Simulator core inlet flow rate is slightly higher than RETRAN. However, the difference is small and is short-lived since the recirculation pumps trip soon after the MSIV closure. Following the recirculation pump trip, the Simulator core flow coastdown is slower and decreases more than RETRAN. This is attributed to the differences in the initial recirculation flow rates and coastdown characteristics modeled in the two codes (see the discussion for Figure 3-2.6). Following full HPCI injection (after 180 seconds), the Simulator predicts a significant increase in core flow as compared to the dampened response calculated by RETRAN. This is attributed to the location of the HPCI injection modeled in the two codes. The Simulator incorrectly models the HPCI injection at the core inlet, thus promoting a forced flow type of response. The HPCI elevation has been correctly modeled in RETRAN at the feedwater sparger location.

Once decay heat levels are reached, the core inlet flow rate stabilizes. This occurs after 1050 seconds in RETRAN and after 1300 seconds in the Simulator, and corresponds to the SLCS injection capacities assumed for the two codes. The Simulator stabilizes at a lower flow rate for reasons discussed under Figure 3-2.6. The second flow swell observed in the Simulator is attributed to the second major actuation of the HPCI system.

Figure 3-2.5 displays Core Inlet Enthalpy. The initial Simulator core inlet enthalpy is slightly higher than RETRAN; however, the difference is negligible to the overall transient. Following recirculation pump trip, the core inlet flow decreases creating degraded core cooling and, consequently, an increase in core inlet enthalpy. The Simulator predicts an injection of the

HPCI system immediately following the MSIV closure caused by a high drywell pressure signal (see Figure 3-2.7). This creates a distortion in the initial enthalpy rise calculated by the Simulator as compared to RETRAN. Otherwise, the two codes match well up to 280 seconds. After 280 seconds the differences in core inlet flow and core power determine the outcome of the core inlet enthalpy where the responses seems to be reasonable.

Once decay heat levels are reached, the core inlet flow rate, core power, and thus the core inlet enthalpy stabilize. This occurs after 1050 seconds in RETRAN and after 1300 seconds in the Simulator, and corresponds to the SLCS injection capacities assumed for the two codes. The two enthalpy swells observed in the Simulator are attributed to the two terminations of the HPCI system.

Figure 3-2.6 displays Recirculation Loop Pump Flow. The initial Simulator recirculation pump flow is significantly higher than RETRAN. The plant is designed to recirculate 50% of the total core flow through the two recirculation loops, where the remaining 50% is the suction flow through the jet pumps. At 100% rated power/flow conditions, this corresponds to 3333 lb/sec per loop. Clearly, the initial Simulator recirculation pump flow rate is much too large. This leads to the conclusion that either the efficiency of the recirculation pumps, or the Simulator form loss coefficient at the jet pump suction, is too high. Since the Simulator core inlet flow (see Figure 3-2.3) stabilizes at a low value relative to RETRAN, the latter of the two conclusion is most likely valid.

Figure 3-2.7 shows HPCI Pump Flow. The Simulator predicts a brief initiation of HPCI shortly after the MSIV closure. This is attributed to the high drywell pressure setpoint being reached that would occur for an ATWS of this severity. The HPCI turbine stop valves close on a receipt of a high reactor water level signal. Once the stop valves close, HPCI remains secured until the signal is cleared by a low water level signal. The initial level swell calculated by RETRAN was large enough to cancel HPCI due to the high drywell pressure signal; where as the Simulator predicted a delayed water level swell thus allowing HPCI to inject briefly.

Both codes predict HPCI initiation on low reactor vessel water level at about the same time. RETRAN models the twenty-five second normal startup to reach full HPCI flow whereas the Simulator does not. The Simulator predicted the full HPCI flow rate as approximately 565 lb/se.. This is significantly lower than the 586.7 lb/sec flow ate modeled in RETRAN. The flow rate in RETRAN is based on 4250 gpm with an upstream pressure of 1200 psia and a temperature of 100 degrees Fahrenheit (this assumes that the condensate storage tank is used as the water source). If the Simulator is using the same volumetric flow rate and upstream pressure, then the source water would be saturated. Therefore, the HPCI water temperature assumed by the Simulator may be incorrect. The second initiation of HPCI in the Simulator is attributed to the second reactor vessel low water level signal (see Figure 3-2.2).

Figure 3-2.8 shows Safety/Relief Valve(s) Flows. This is actually the steam discharge to the T-quenchers in the PSP by the four pressure relief valves, not including the steam discharge through the two safety valves. Both codes predict the initial automatic opening of all four valves, as well as lifting of the two ASME Code Safety valves. The response of the Simulator is in good agreement with RETRAN. The Simulator seems to have slightly lower relief valve closure setpoints as compared to those modeled in RETRAN; however, the differences are small and do not significantly affect the transient simulation.

Figure 3-2.9 depicts the Core Void Fraction predicted by RETRAN and the Simulator, respectively. The RETRAN predictions are more pronounced than those calculated by the Simulator. This is attributed to the differences in the way the bypass region is modeled. RETRAN models the bypass separately from the fuel region of the core. The impact of modeling of the bypass and core as a single entity is detailed in discussion of the results for the previous section.

3.3 RPT and ATWS Benchmark Conclusions

The overall predicted responses of the Simulator for the ATWS and RPT comparisons are generally quite good. However, based on the observations of the Simulator compared actual plant data (for the RPT simulation) to RETRAN (for the ATWS simulation) and there are five areas where the Simulator could use improvements. These are: 1) Modeling the HPCI location at the feedwater elevation as flow through spargers into the reactor vessel downcomer region instead of at the core inlet; 2) Adjusting the form loss coefficient at the jet pump suction to achieve 50% flow at 100% flow conditions instead of the present 35% flow; 3) Model the HPCI subcooled liquid at the Condensate Storage Tank temperature instead of at 212 degrees Fahrenheit; 4) Modeling a separate volume for the bypass region rather than combining it with the fuel region; and 5) Modeling the M/G coastdown characteristics more precisely. By improving these five areas, it is believed that the Simulators capability to predict plant response more accurately would increase.

TABLE 3-1.1

VY RPT Initial Conditions

Parameter	Plant-Data	Simulator	Note
Core Thermal Power (P/P_o)	0.83	0.84	A
Dome Pressure (psia)	1000	1000	
Narrow Range Water Level (inches) (above top of active fuel)	151.8	151.7	
Core Inlet + Bypass Flow (lb/sec)	11309	11217	B
Main Steam Flow (lb/sec)	1429	1421	C
Feedwater Temperature (^oF)	359	355	
M/G Set Speed (Ω/Ω_o)	0.79	0.74	D
Exposure State-Point	EOFPL-08	EOFPL-12	

(A) Rated core power equals 1593 MWth.

(B) Rated core flow equals 13333 lb/sec.

(C) Rated main steam/feedwater flow equals 1786 lb/sec.

(D) Rated M/G speed equals 1120 RPM

TABLE 3-2.1

VY ATWS Initial Conditions

Parameter	RETRAN	Simulator	Note
Core Thermal Power (P/P_o)	1.00	1.00	A
Dome Pressure (psia)	1020	1020	
Wide Range Level (inches) (above vessel zero)	510	510	
Narrow Range Level (inches) (above top of active fuel)	158.5	158.5	
Core Inlet + Bypass Flow (lb/sec)	13333	13659	B
Bypass Flow (lb/sec)	1448	NA	C
Core Inlet Enthalpy (Btu/lb)	520	522	
Recirculation Pump Flow (lb/sec)	3260	4392	D
HPCI Flow (lb/sec)	0.0	0.0	
S/RV Flow (lb/sec)	0.0	0.0	
Main Steam Flow (lb/sec)	1786	1717	E
Feedwater Flow (lb/sec)	1786	1707	E
M/G Speed (Ω/Ω_o)	0.94	0.94	F
SLCS Capacity (gpm) (boron injection)	43	35	G
Exposure State-Point	EOFPL-11	EOFPL-12	

(A) Rated core power equals 1593 MWth.

(B) Rated core flow equals 13333 lb/sec.

(C) The Simulator does not have a bypass model, the core inlet for the Simulator includes the bypass flow.

(D) Rated recirculation flow equals 3333 lb/sec per loop.

(E) Rated main steam/feedwater flow equals 1786 lb/sec.

(F) Rated M/G speed equals 1120 rpm.

(G) 35 gpm corresponds to the technical specification value, 43 gpm corresponds to the maximum capacity of the pump.

TABLE 3-2.2

VY ATWS RETRAN Trips and Operator Actions

Trips

1 Recirculation Pump Trip on high pressure.

2 Feedwater Pump Trip on high water level.

Operator Actions

1 Override the Automatic Depressurization System.

2 Initiate the Standby Liquid Control at two minutes plus a 30-second delay to sweep the injection lines.

3 Disable High Pressure Coolant Injection and Reactor Core Injection Coolant suction shift from the control storage tank to the Pressure Suppression Pool on high Pressure Suppression Pool water level.

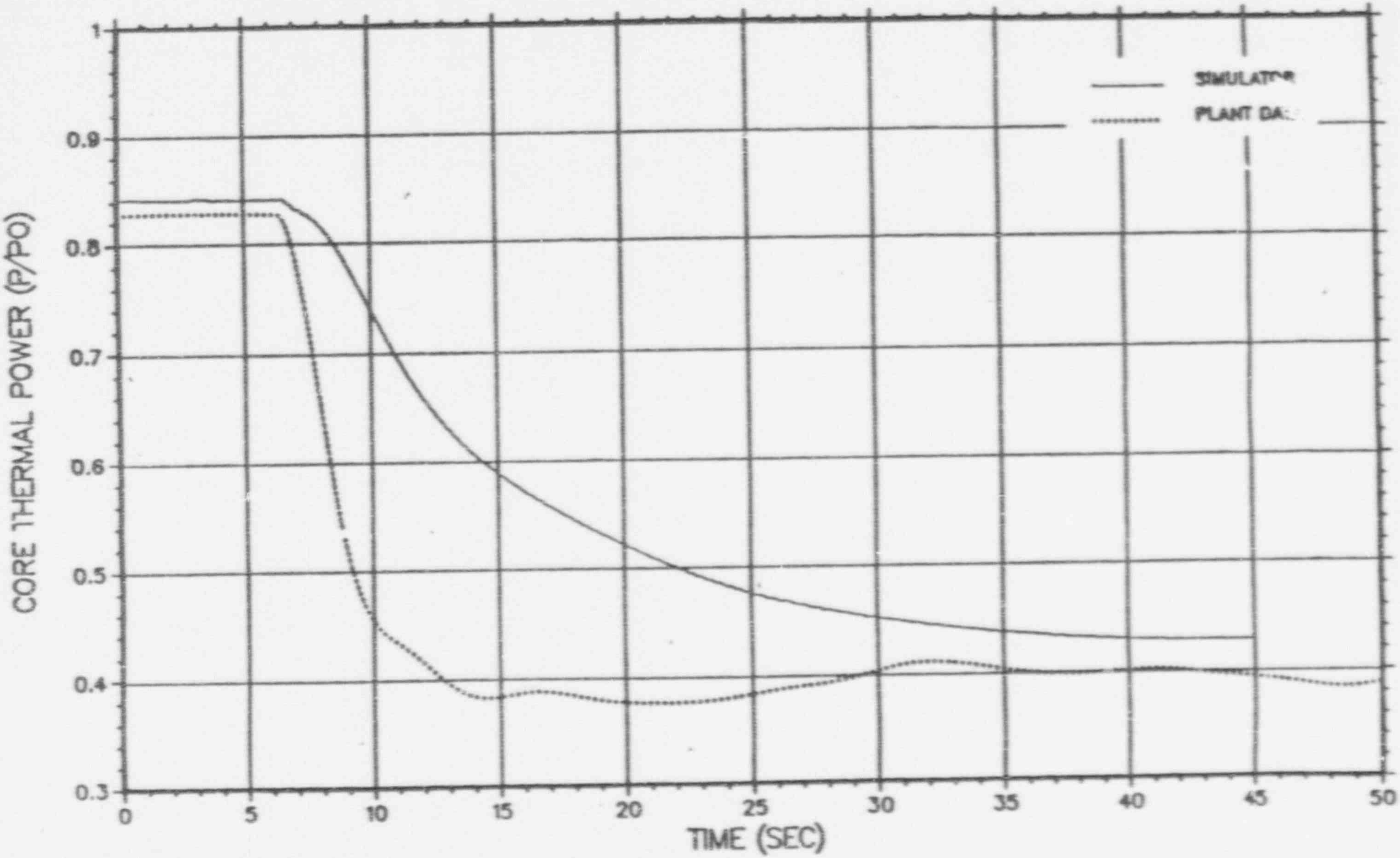


Figure 3-1.1 VY RPT Core Thermal Power

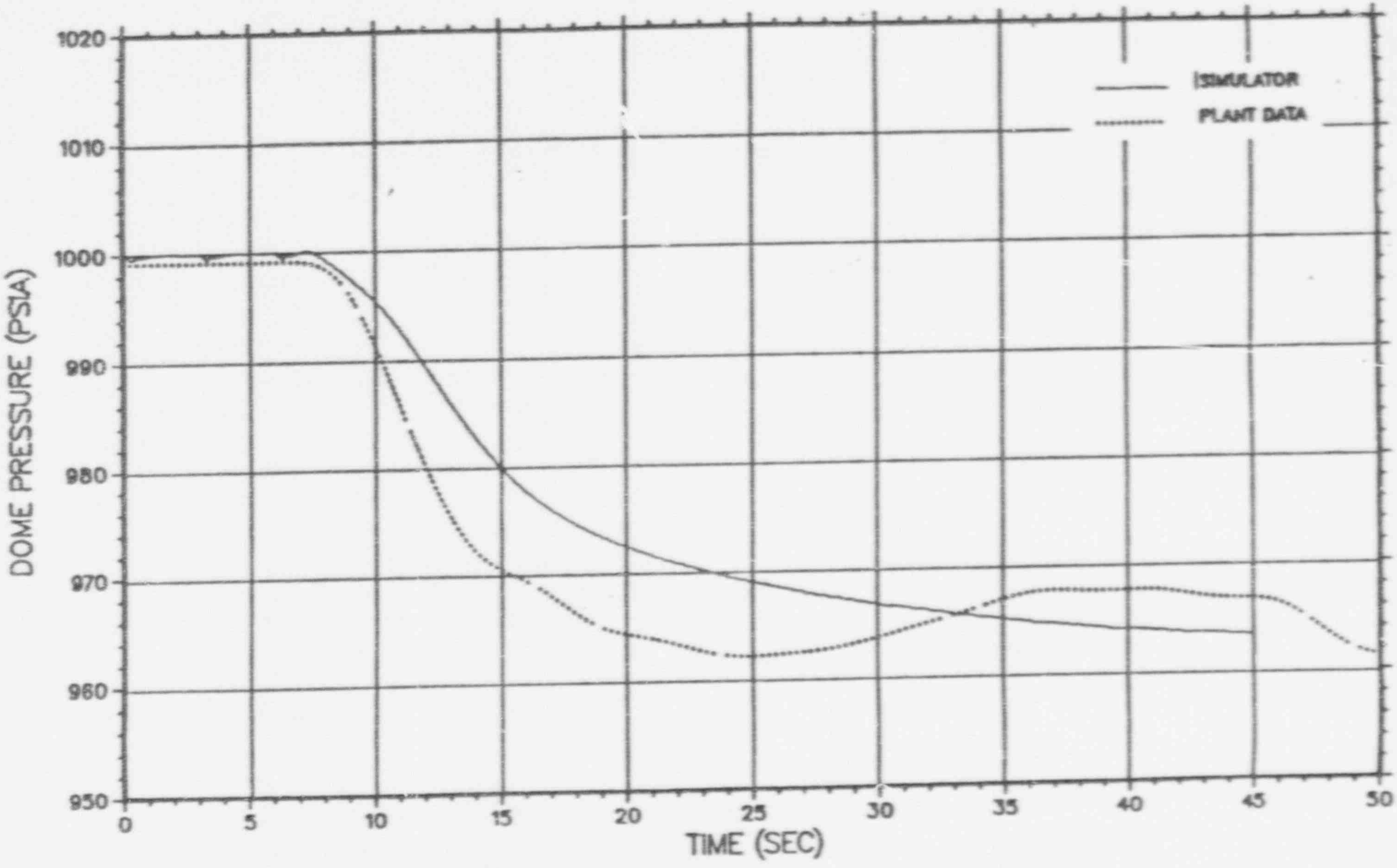


Figure 3-1.2 VY RPT Reactor Vessel Steam Dome Pressure

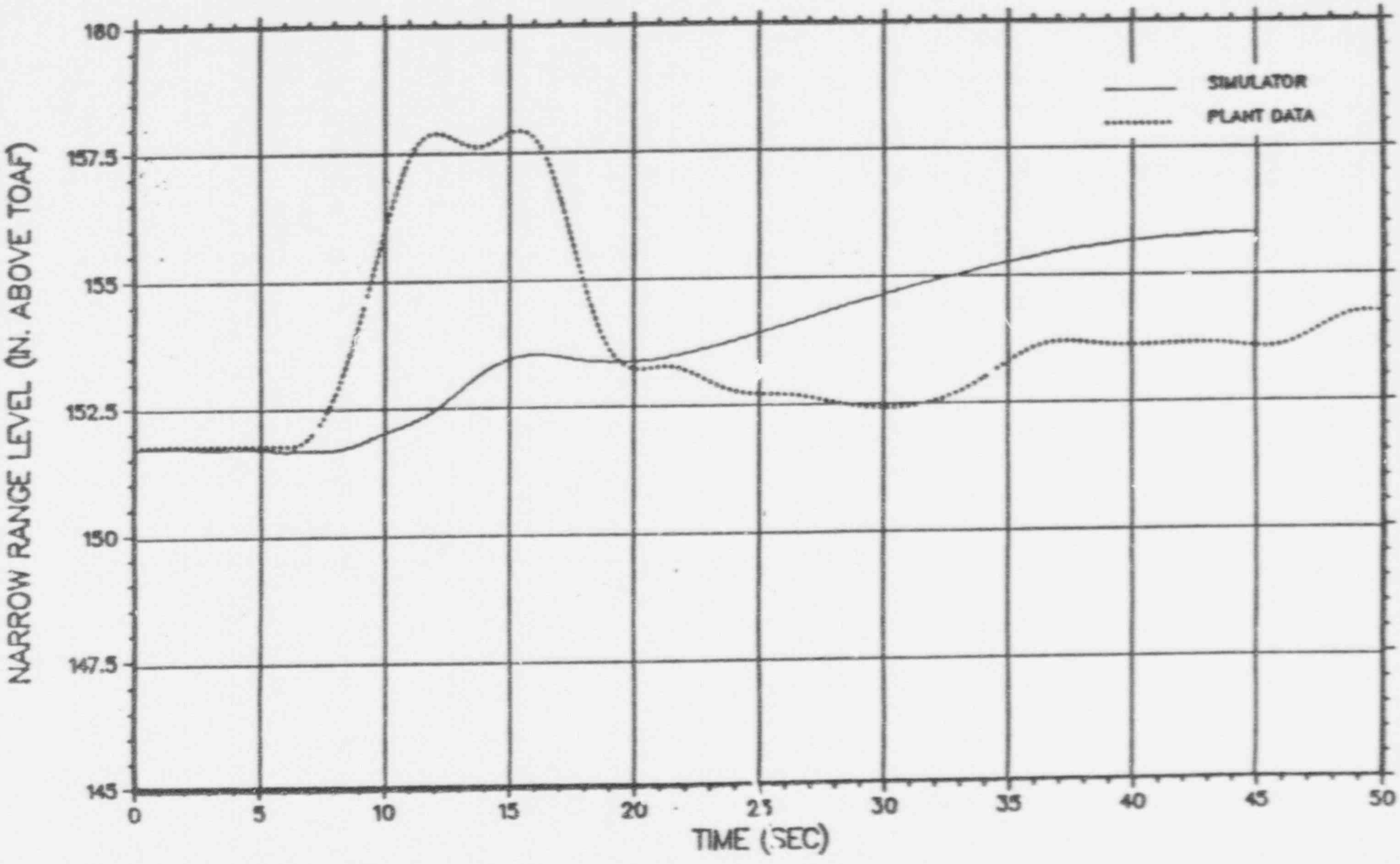


Figure 3-1.3 VY RPT Reactor Vessel Narrow Range Water Level

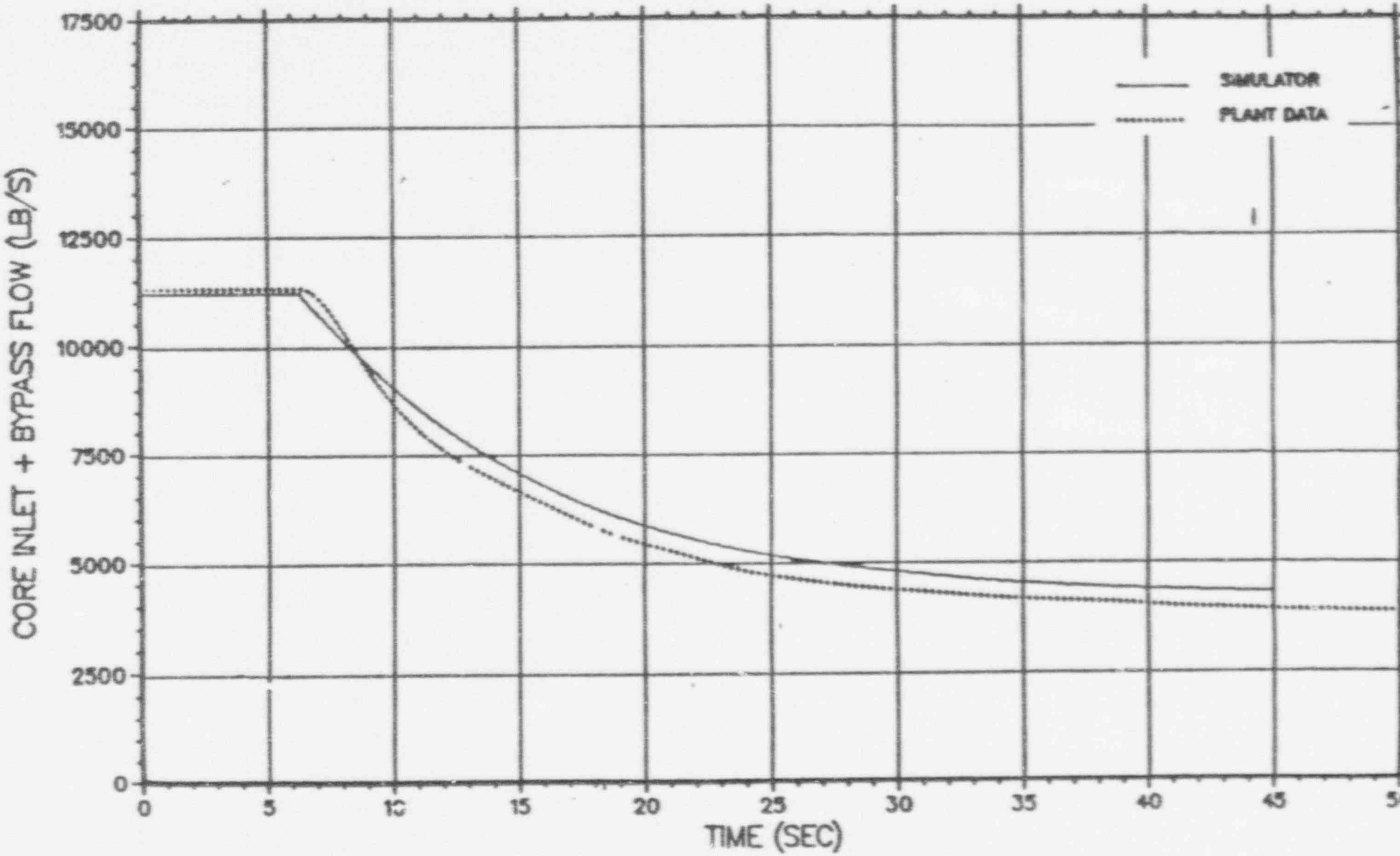


Figure 3-1.4 VT RPT Core Inlet and Bypass Flow

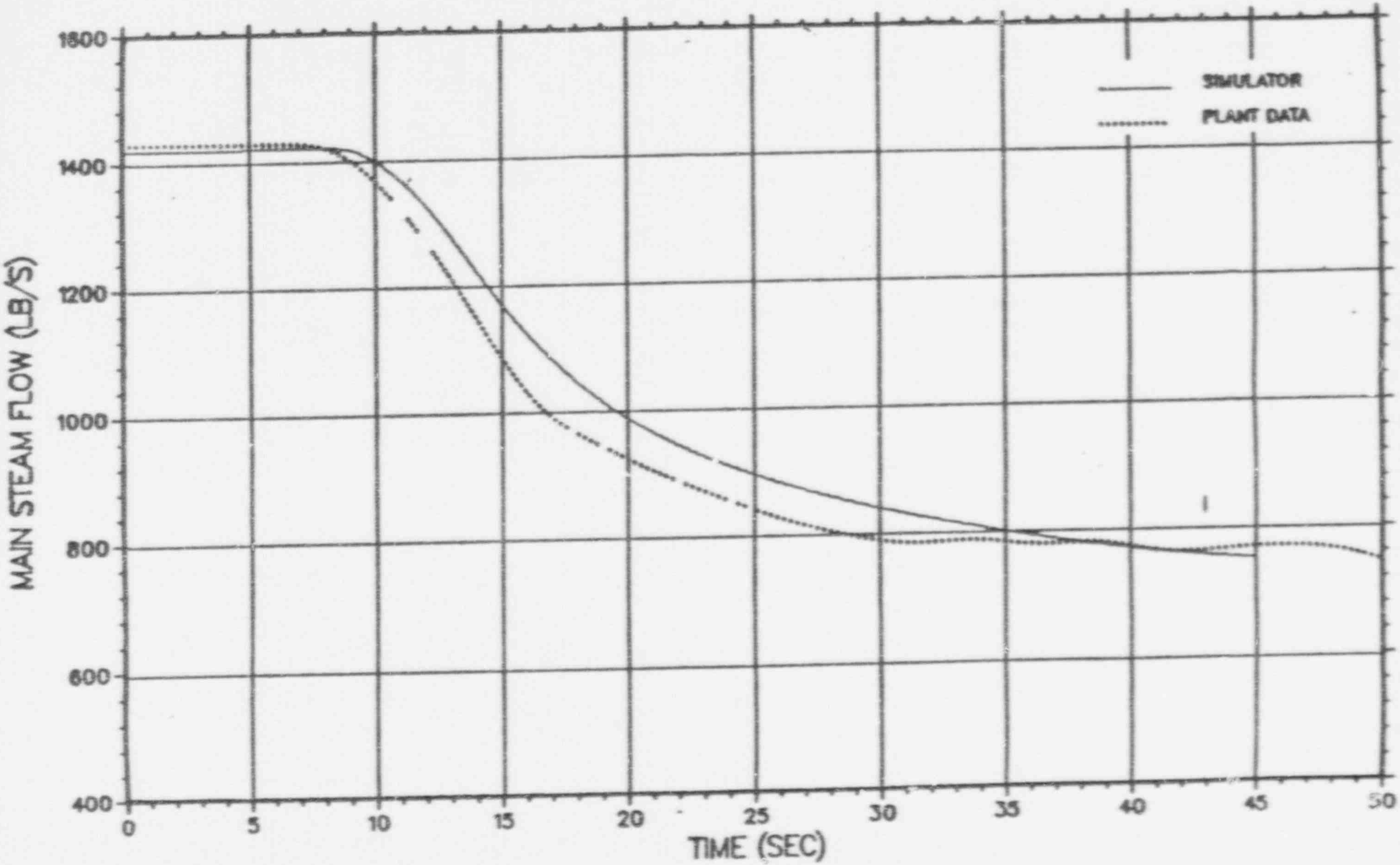


Figure 3-1.5 TT RPT Main Steam Line Flow

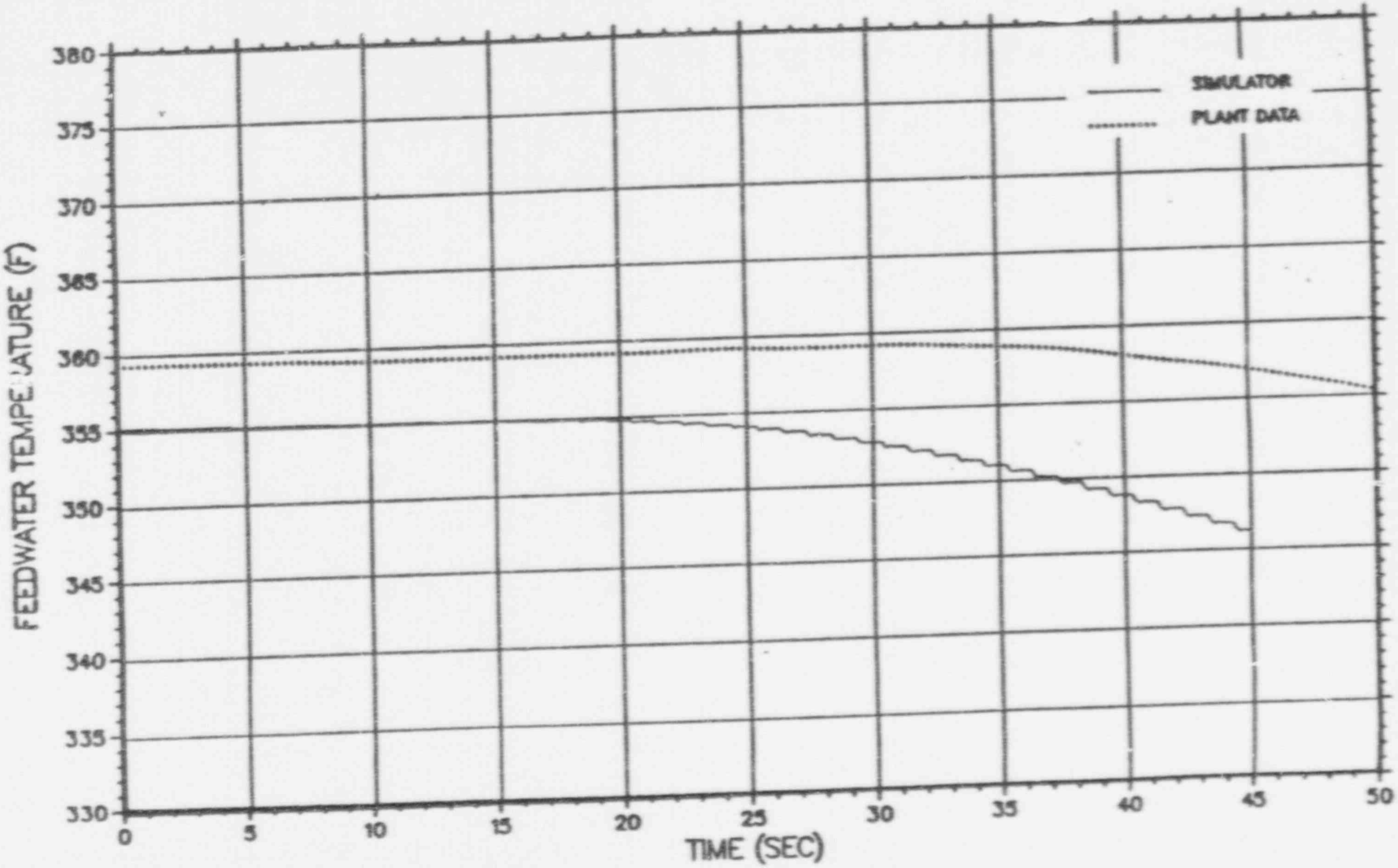


Figure 3-1.6 VT RPT Feedwater Temperature

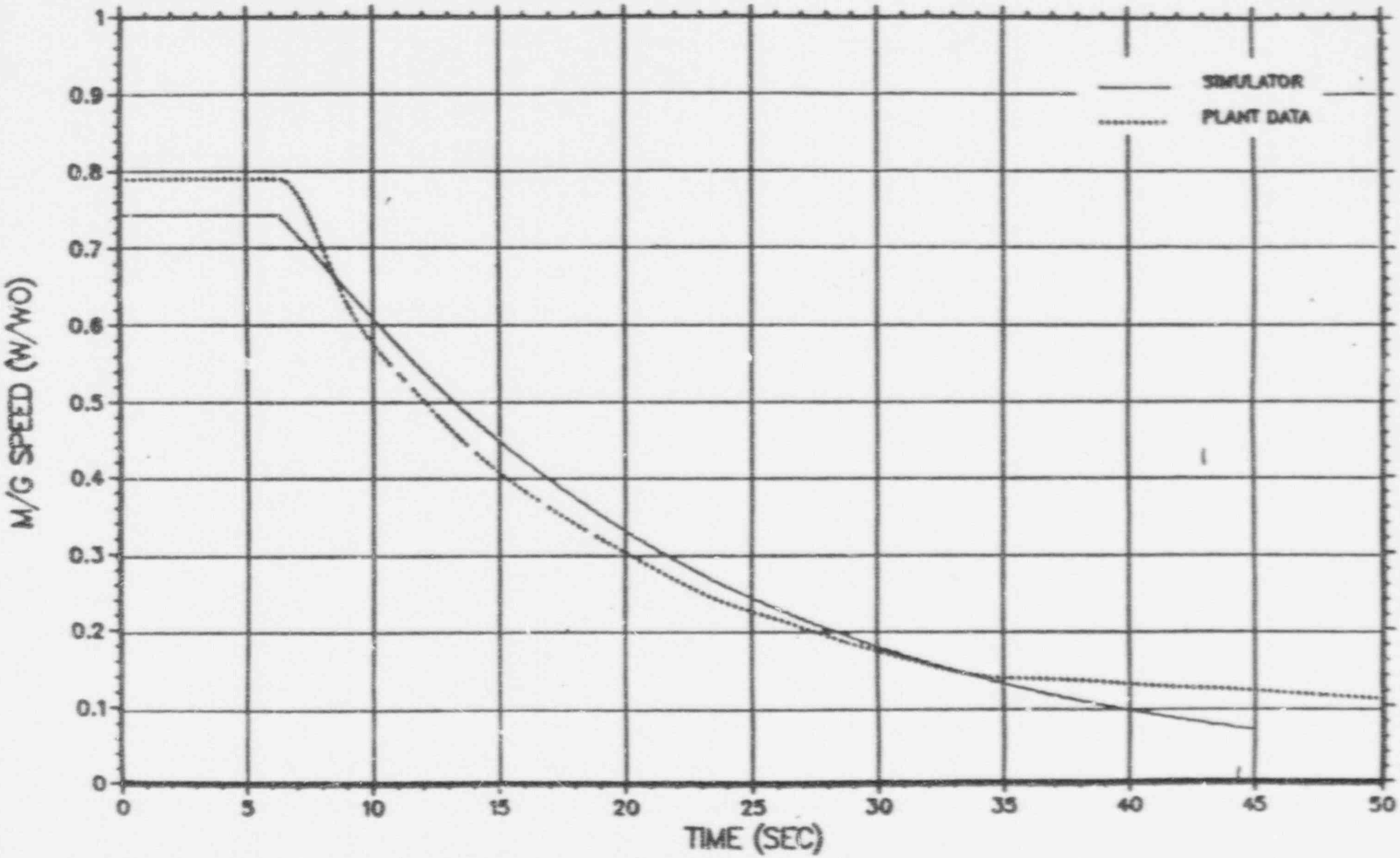


Figure 3-1.7 VT RPT Reactor Recirculation Loop M/G Set Speed

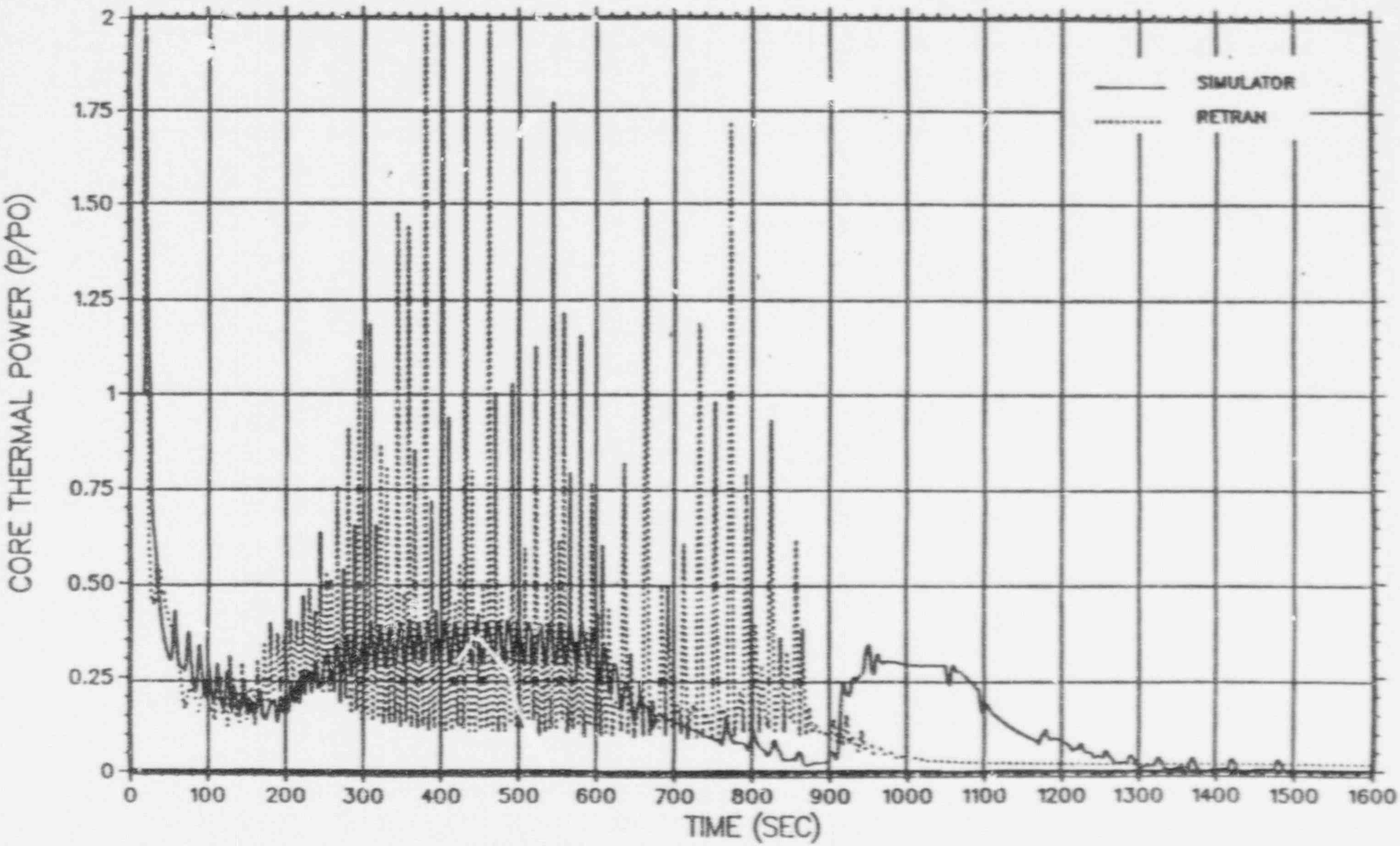


Figure 3-2.1 VT ATWS Core Thermal Power

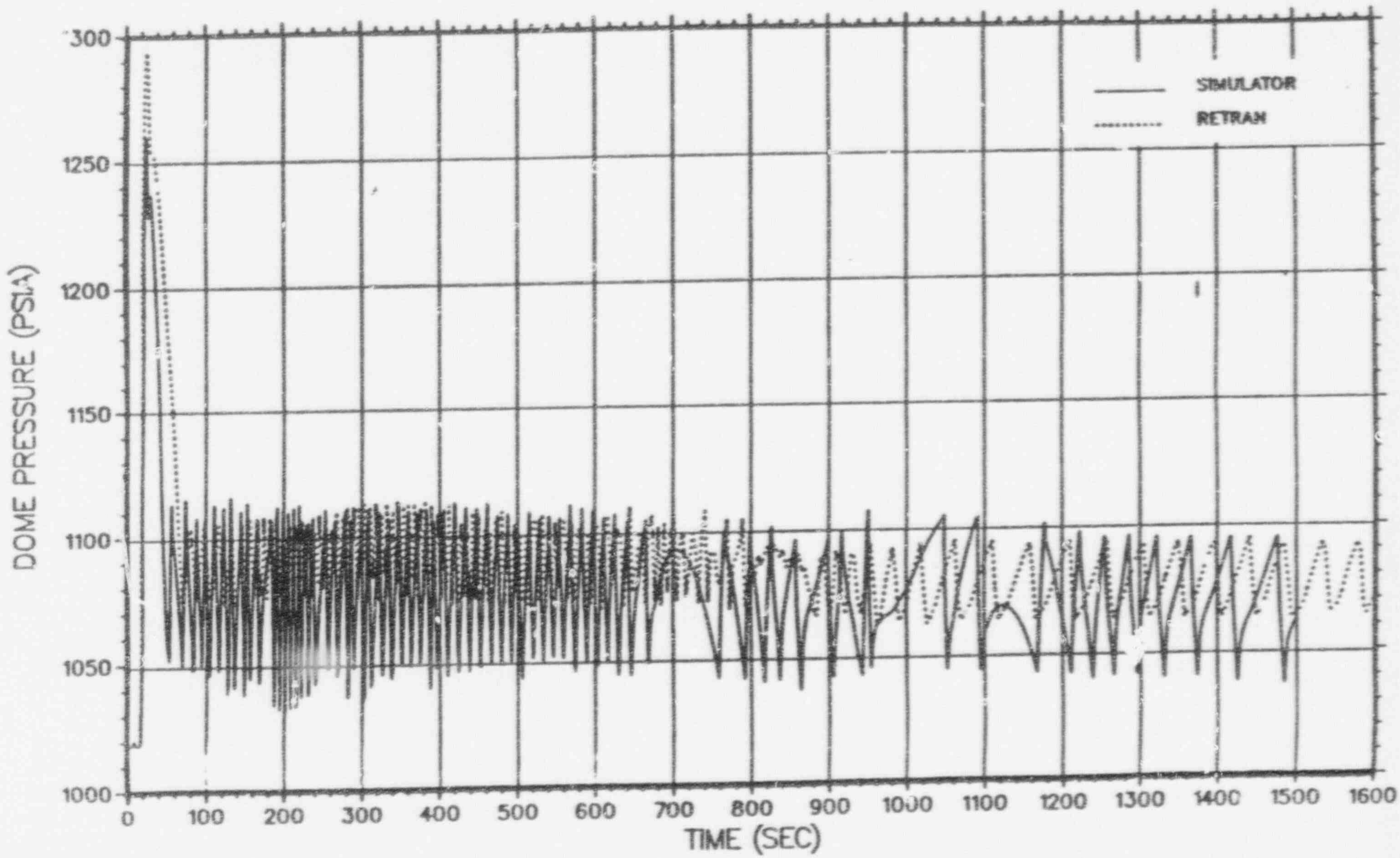


Figure 3-2.2 VY ATWS Reactor Vessel Steam Dome Pressure

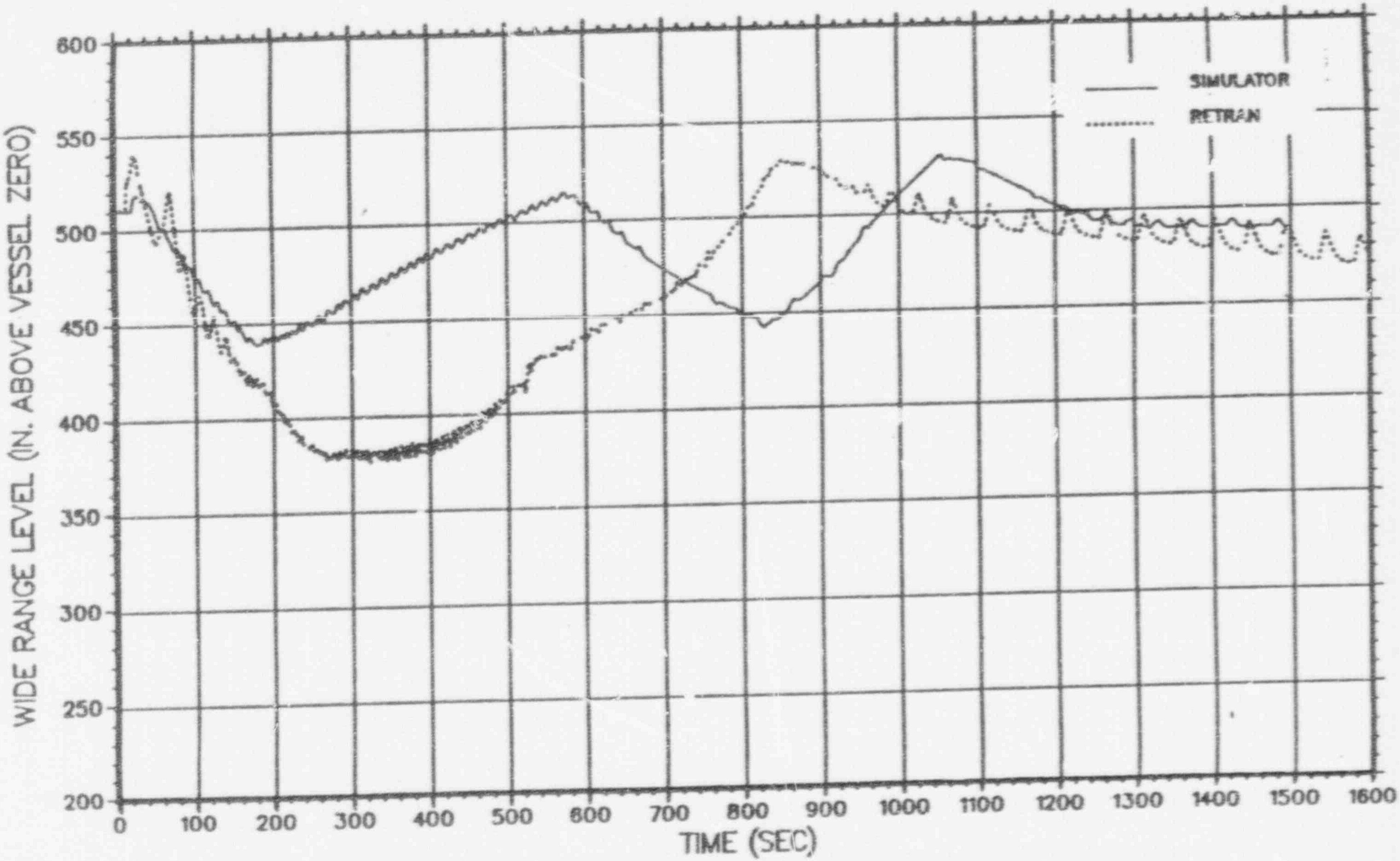


Figure 3-?.3 VY ATWS Reactor Vessel Wide Range Water Level

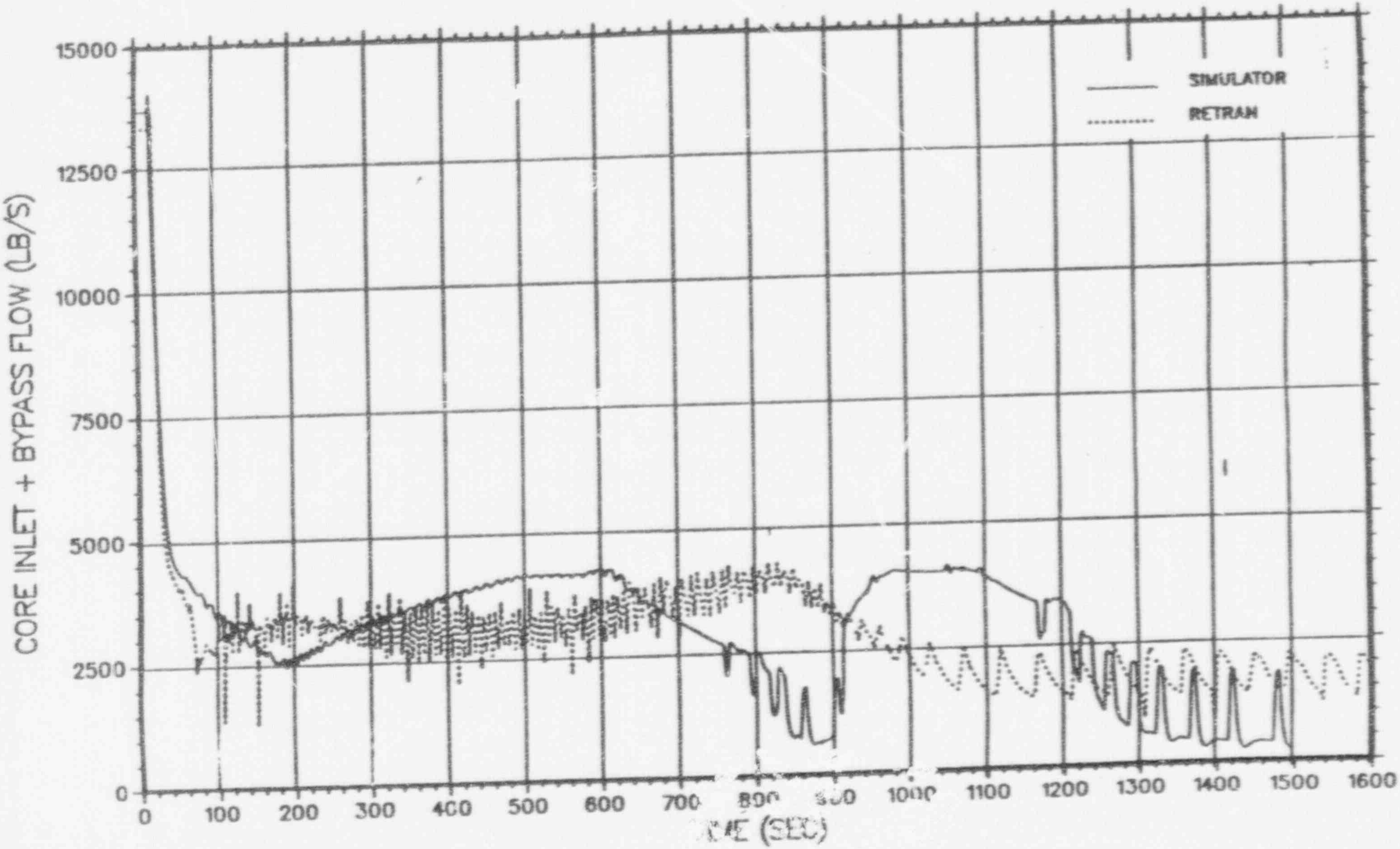


Figure 3-2.4 VT ATWS Core Inlet and Bypass Flow

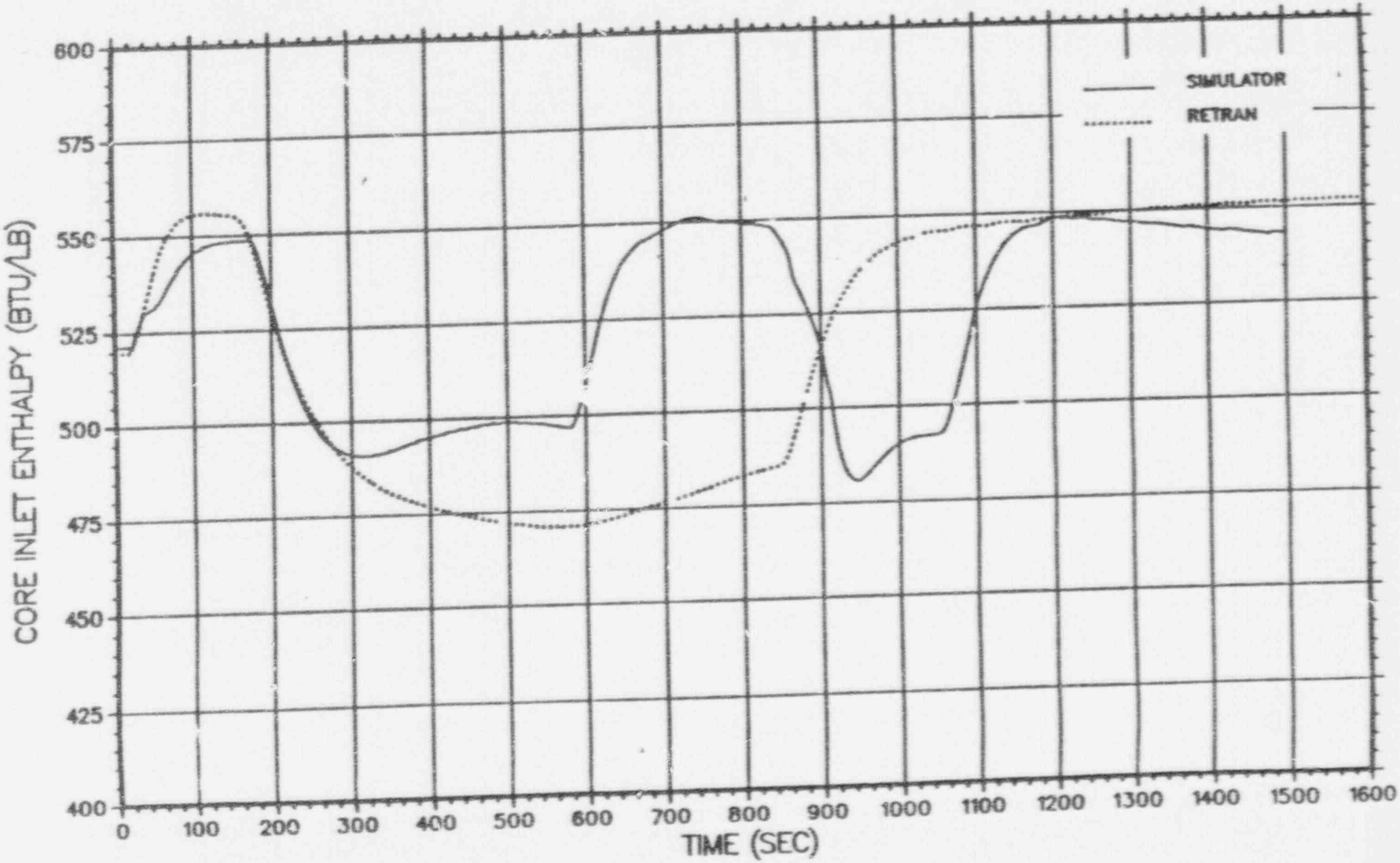


Figure 3-2.5 VY ATWS Core Inlet Enthalpy

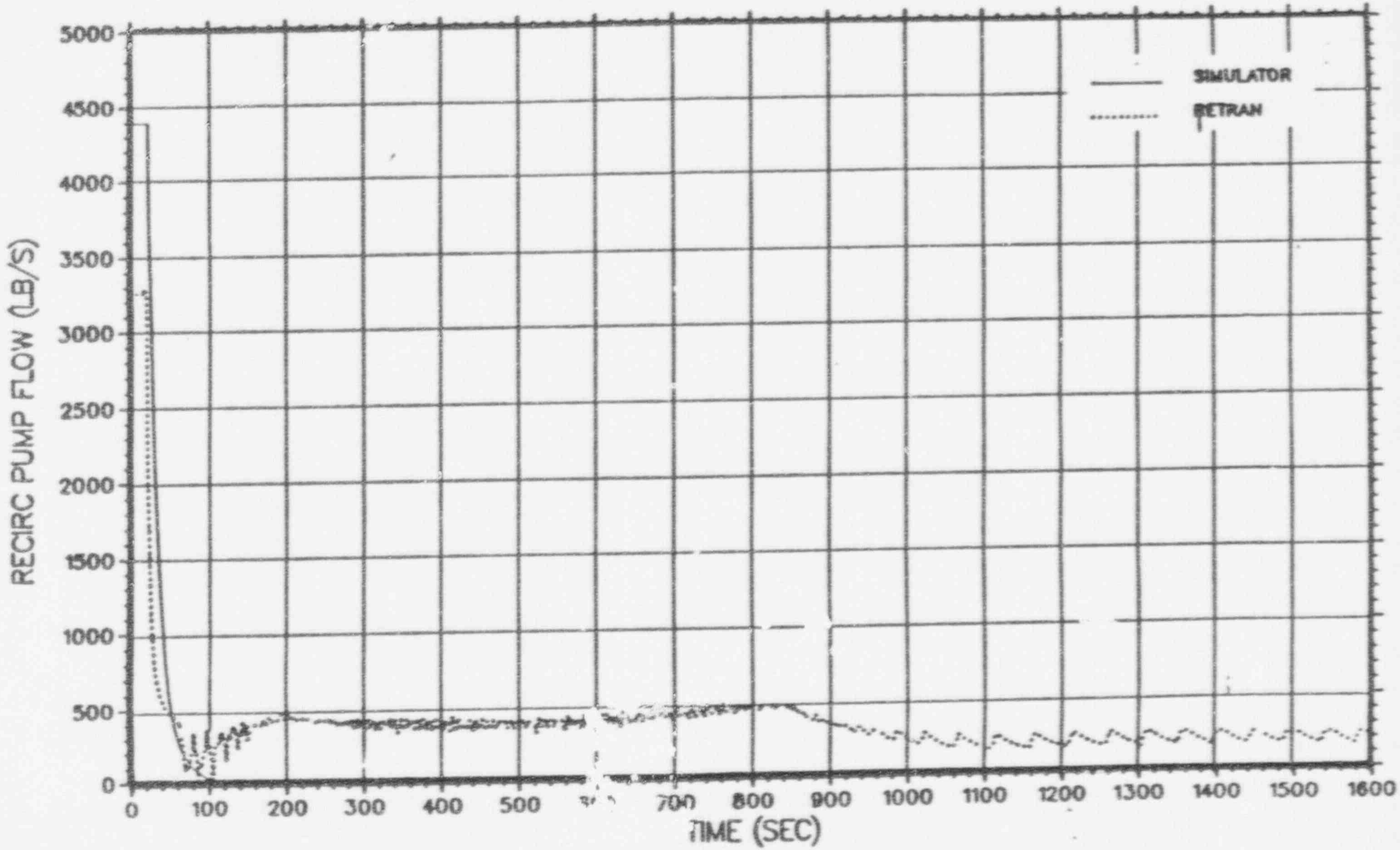


Figure 3-2.6 9. TMI Reactor Recirculation Loop Pump Flow

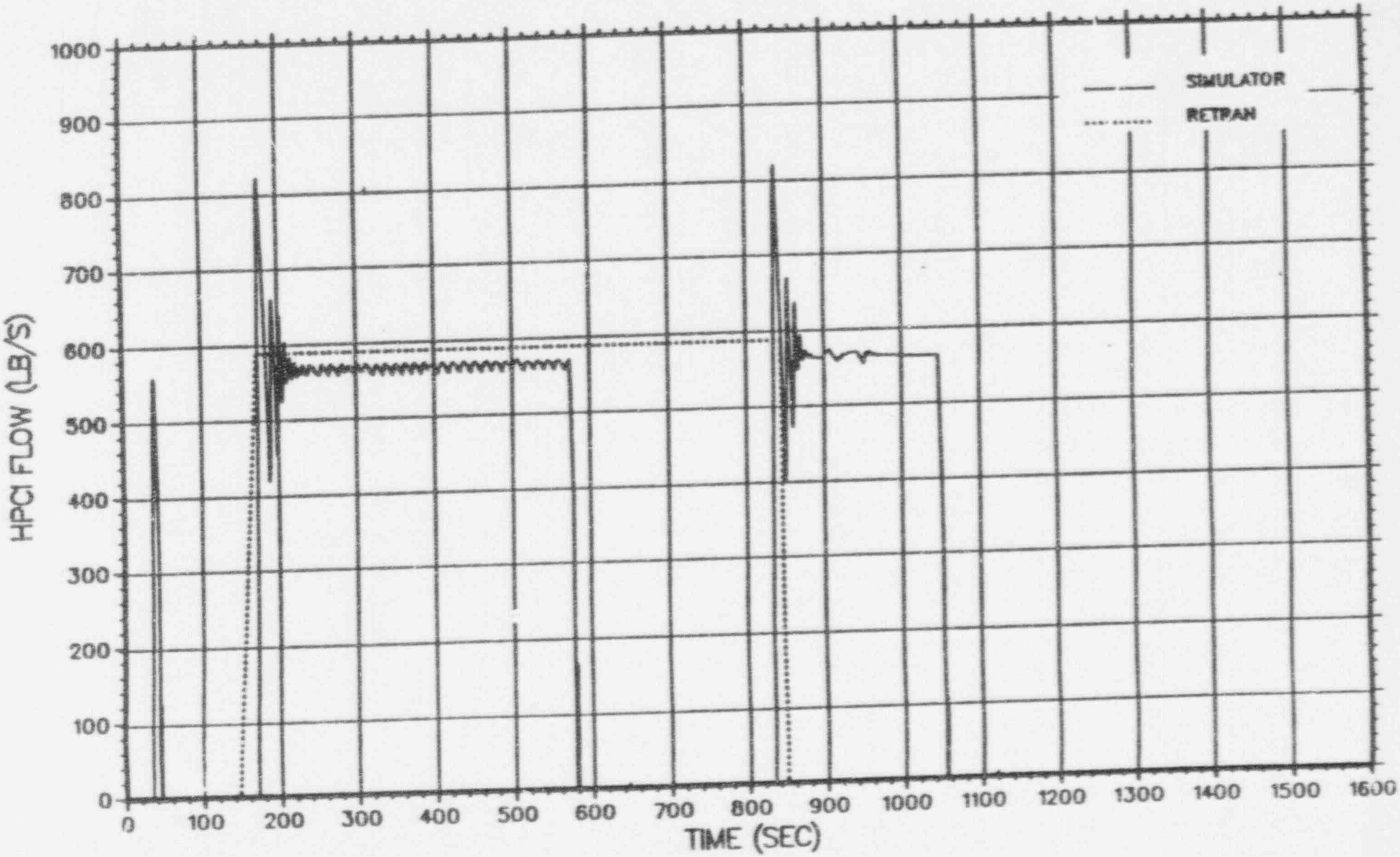


Figure 3-2.7 VY ATWS HPCI Pump Flow

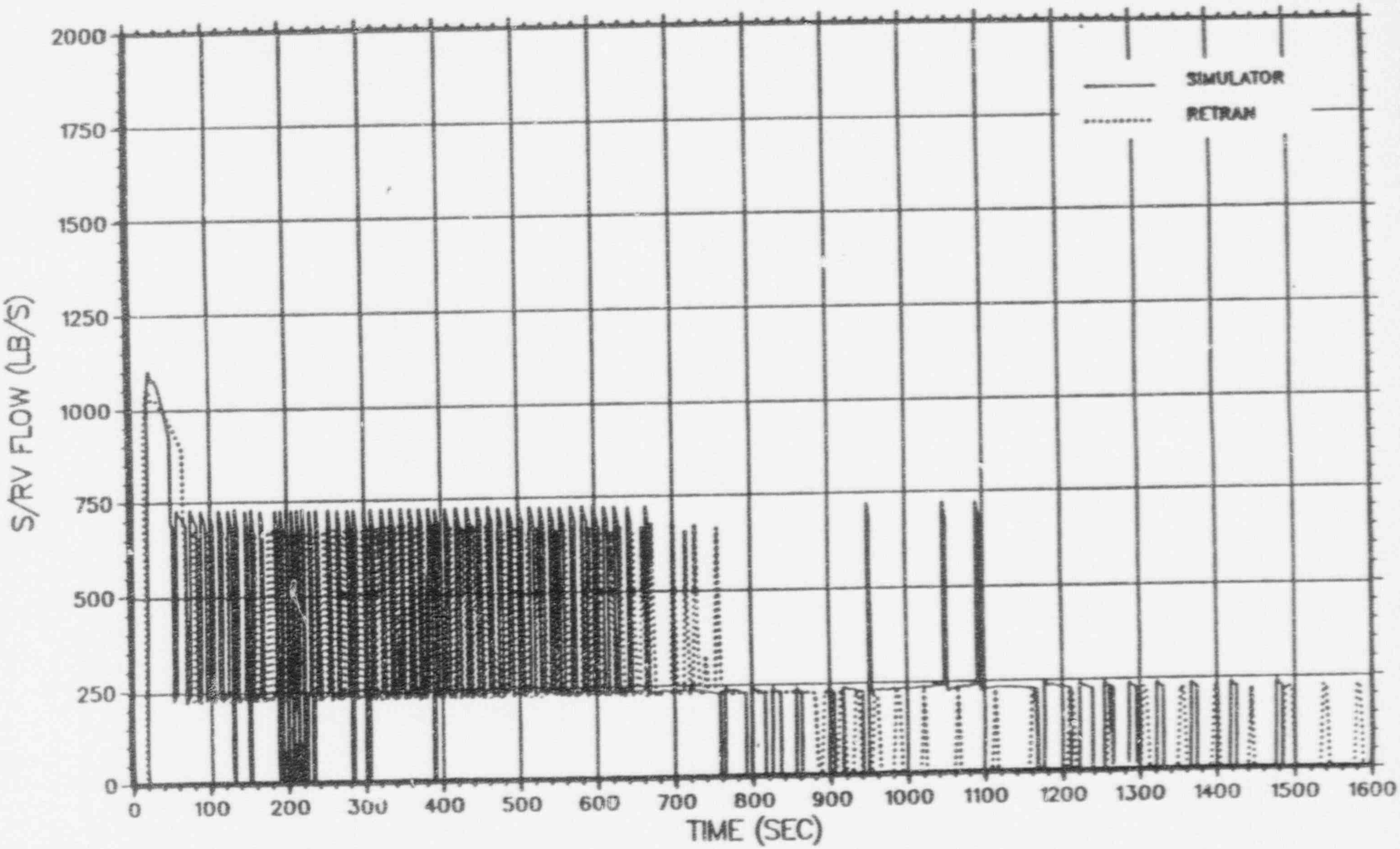
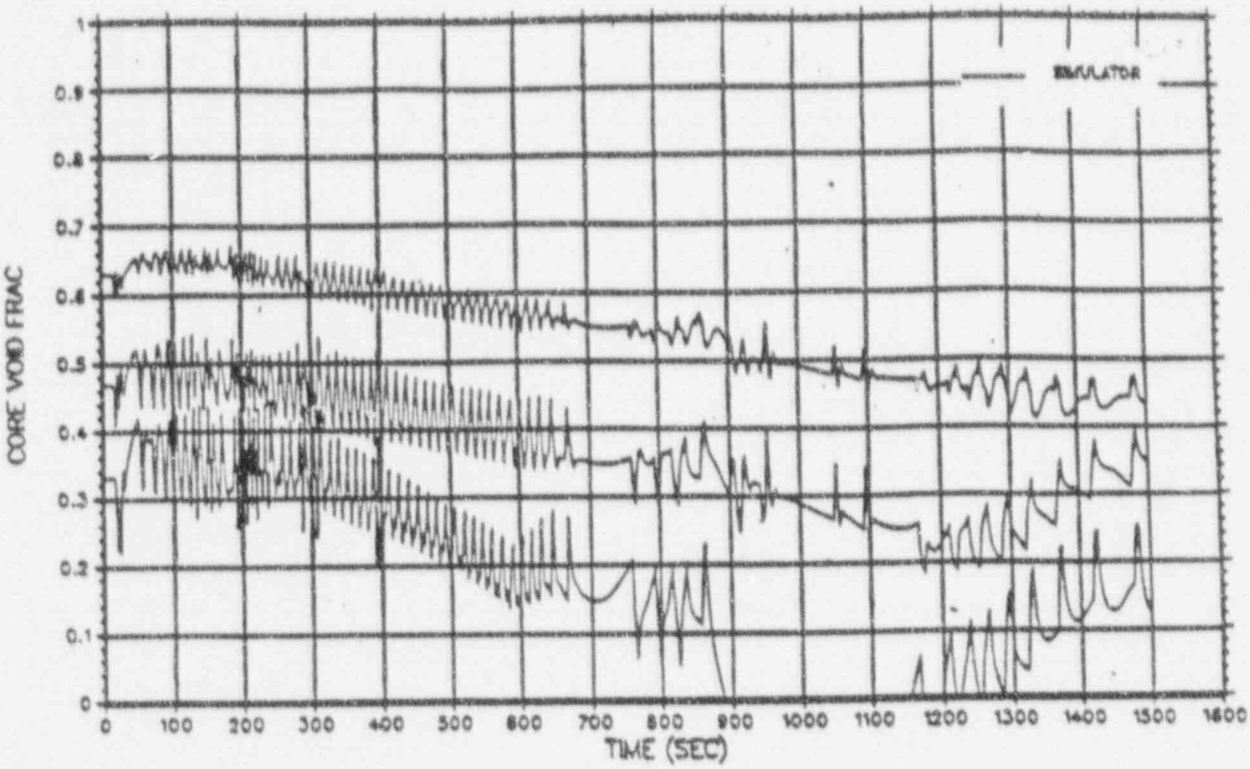


Figure 3-2.8 VY ATWS Safety/Relief Valve(s) Flow



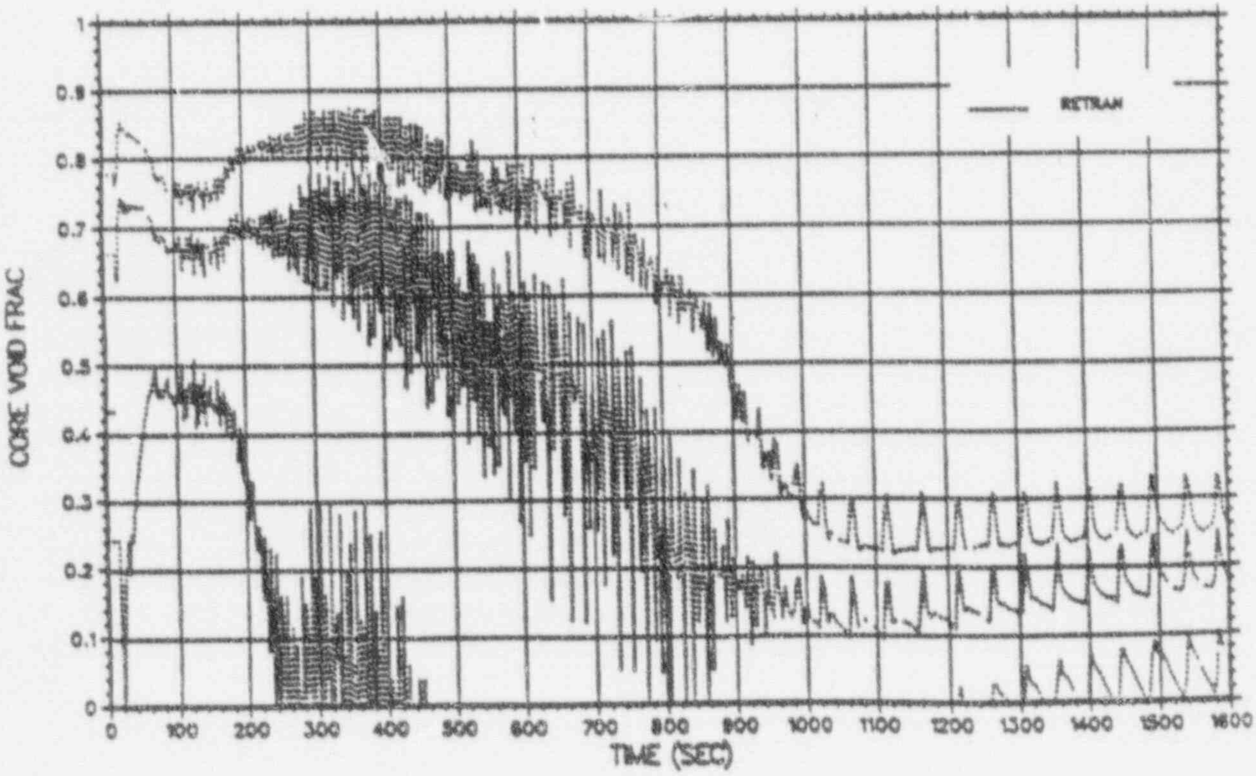


Figure 3-2.9 VY ATWS Core Void Fraction

4.0 LOSS OF COOLANT ACCIDENTS

Three types of loss of coolant accidents were used as benchmark scenarios. They are a classic large break LOCA, a small break LOCA, and a Main Steam Line break outside containment. The first two involve breaks in piping that is carrying initially subcooled fluid, while the later involves a break in a pipe with high quality steam. The RELAP5YA code was employed as the engineering model in all of the LOCA cases.

4.1 INITIAL CONDITIONS

The Large and Small Break LOCA cases were run with the same IC-9 Simulator initial conditions. Likewise, these two LOCA cases were run with the same RELAP5YA initial conditions. The Simulator and RELAP5YA initial conditions are compared in Table 4-2.1. The two sets are close but not identical. Both the Simulator and RELAP5YA used 100% power. However, the Simulator had a Beginning-of-Cycle (BOC) axial profile while RELAP5YA used a chopped cosine axial profile. Initial steam dome pressure and water levels were nearly the same. The initial core flow was 96% of rated for the Simulator and 100% for RELAP5YA.

The Main Steam Line Break case was run with the IC-41 Simulator initial condition. These conditions are discussed in Section 4.4.

4.2 LARGE BREAK LOCA

The Large Break LOCA case is an instantaneous Double-Ended Guillotine (DEG) break in the suction pipe of one recirculation loop. The Simulator and RELAP5YA calculations were run until adequate core cooling was re-established. During the post-test analysis, 25 parameters were compared for this case. Primary emphasis was to ascertain that the Simulator response showed proper parametric trends and significant event times.

4.2.1 LB-LOCA Accident Assumptions

The large break LOCA case was initiated from rated conditions. The accident assumed a double-ended guillotine break occurred in a suction pipe upstream from the recirculation pump, with a coincident loss of off-site power. The accident assumptions are summarized in Table 4-2.2. The major assumptions include loss of auxiliary power, failure of the LPCI-B injection valve, and failure of the RCIC steam turbine admission valve. Thus, the available systems for injecting water to the reactor vessel are:

a. HPCI from Condensate Storage Tank (CST)

b. 2 - Low Pressure Core Spray (LPCS) systems from the Torus

c. 1 - Low Pressure Coolant Injection (LPCI) system from the Torus

The nodalization diagrams for the VY-NSSS and containment are described in Appendix A.2. This integrated model contains 169 volumes, 186 junctions, and 176 heat structures. These represent the NSSS to the Turbine Stop Valves (TSVs), and the containment Drywell (DW), Vent System (VS), and Wetwell (WW). The NSSS and containment models are coupled through the hydraulic response associated with the pipe break, S/RVs, and ECCS injection, and the emergency safeguard instrumentation responses in these regions.

4.2.2 LB-LOCA Comparison

Table 4-2.3 summarizes the timing of significant events for this case. Generally, the Simulator and RELAP5YA significant times are identical or within several seconds of one another. The notable exception is that the Simulator shows a much later core heatup and subsequent cooldown. The Simulator and RELAP5YA results are compared in Figures 4-2.1 through 4-2.20.

Figure 4-2.1 compares the Simulator and RELAP5YA power histories. Both show a rapid decline in power production, due to rapid core voiding and SCRAM, down to very similar decay heat levels. SCRAM is initiated by the High Drywell Pressure trip.

The NSSS steam dome pressure history from the Simulator and RELAP5YA calculation are compared in Figure 4-2.2. These show similar trends. Differences in depressurization rates were traced to the following:

a. Differences in the Steam Line flow rates out the Turbine Bypass Valves prior to MSIV closure at eight seconds.

b. Differences in the steam qualities out the break locations until 40 seconds.

The Simulator's vessel pressure history is acceptable since it's display in the Control Room should elicit the proper response from the operators for this event.

Figure 4-2.3 shows the narrow-range level histories. The Simulator and RELAP5YA narrow-range levels drop to 77", the minimum displayed value, at nearly the same time. The Simulator's internally calculated value continues to drop to zero, but the Simulator's Control Room displayed value is limited to 77". The two RELAP5YA values, actual and indicated, and the Simulator's value are in general agreement for this accident.

Figure 4-2.4 shows the wide-range level histories. The single Simulator level is compared to three RELAP5YA levels, corresponding to the outer shroud, inner shroud and the core bypass regions. The plant instrumentation response will reflect the maximum value between the outer and inner shroud regions. From 0 to 24 seconds, the Simulator level follows the decrease predicted by RELAP5YA, but declines slower since the Simulator break flow rates are somewhat smaller. After 24 seconds when LPCI and LPCS injection begin, the Simulator level recovers faster even though the combined ECCS flow rates are less than RELAP5YA. Therefore, we infer that the vessel and core nodalization, the ECCS injection locations, and the water level calcualtions in the Simulator need further improvement to provide a more realistic simulation for this large suction break case.

Figure 4-2.5 shows feedwater flow rate histories. The RELAP5YA calculation assumes this flow ramps to zero in five seconds. The Simulator feedwater response was nearly the same.

Figure 4-2.6 compares the main steam line flow rate histories. The general trends are about the same. First, the flow decreases when the stop valve closes; then, the flow increases when the bypass valves open. Finally, the steam line flow is stopped completely by the MSIV closure. The Simulator flow rates during the turbine bypass period are more realistic since the Simulator has a more complete model of the Turbine and Bypass Control System at this time.

Figure 4-2.7 shows upstream and downstream break flows. The Simulator upstream and downstream break flows are identical during the subcooled blowdown period. These flows then decline rapidly to values that correspond to steam flow when the calculated liquid level for each flow path reaches the break location. In contrast, the RELAP5YA upstream and downstream break flows are unequal, because the two phase conditions feeding each path are different. The Simulator does not account for constraints imposed on the upstream and downstream flows by geometry in the recirculation loops, whereas the RELAP5YA model does. Specifically, the downstream break flow is limited by ten jet pump drive nozzle flow areas; the upstream flow area is identical to the suction pipe area.

The RELAP5YA break flows decay smoothly, whereas the Simulator downstream break flow drops sharply near 8 seconds and the upstream break flow drops sharply at 21 seconds. Despite the two unnatural drops, the sum of the Simulator break flows are a fair approximation of the sum of RELAP5YA break flows up to 11 seconds. From 11 to 30 seconds, the total RELAP5YA break flow exceeds the total Simulator break flow. Beyond 30 seconds, the total break flows become similar again. The Simulator break flow calculations were sufficiently realistic for observable parameters, such as system pressure, to be realistic.

Figure 4-2.8 shows the recirculation pump speed histories. Two RELAP5YA histories are shown, one for the Loop A (broken loop) pump and one for the Loop B (intact loop) pump. The Simulator calculates the same speed for both pumps. The intact loop pump speed calculated by the Simulator and RELAP5YA agree very well. However, RELAP5YA calculates a negative overspeed for the broken loop pump after the field breaker trip occurs at 17 seconds; this is realistic since the break is on the suction side of the Loop A pump. In contrast, the Simulator either did not properly model the field breaker trip or did not properly model the broken loop recirculation pump; therefore, the negative overspeed did not occur. Beyond 70 seconds, the pumps rotate very little in both the Simulator and RELAP5YA calculations.

Figure 4-2.9 shows the recirculation loop flow rate histories. Again, the Simulator shows only an intact loop behavior for both loops. The RELAP5YA intact loop flow simply coasts down. The RELAP5YA broken loop flow is negative after the break occurs, and then reaches essentially zero by 60 seconds.

Figure 4-2.10 shows the jet pump flow rate histories. Once again, the Simulator calculation for both loops is consistent with the RELAP5YA intact loop calculation. The Simulator does not calculate the proper response for the broken loop.

The RELAP5YA downstream break flow rate in Figure 4-2.7, broken loop recirculation flow rate in Figure 4-2.9, and broken loop jet pump flow rate in Figure 4-2.10 all show consistent negative (reversed) flow rates during the first 30 seconds of this accident;

the negative peak flow rate is approximately 4200 lbm/sec. In contrast, the Simulator downstream break flow rate peaks at about 13,200 lbm/sec, and is inconsistent with the recirculation loop and jet pump flow rates calculated by the Simulator.

Figure 4-2.11 shows the core inlet mass flow rate histories. The combined core and bypass flow rate, as well as the core only flow rate, are shown for RELAP5YA. The Simulator does not have a separate bypass region, so the core and bypass flow rates are lumped together in the Simulator calculation. The RELAP5YA flows coastdown to low positive values over about 30 seconds, and become somewhat erratic beyond during ECCS injection. The Simulator core flow declines to zero at 8 seconds, which follows the feedwater and jet pump flow coastdowns to zero values in 8 seconds also. This is indicative of certain simplifying assumptions inherent in the Simulator's thermal-hydraulic models that allow faster computation, but at the expense of transient simulation fidelity.

Figure 4-2.12 compares the HPCI flow rate histories. HPCI injection began at 11 seconds in the Simulator and at 20 seconds in RELAP5YA calculations (curve with circles). The Simulator's HPCI injection curve begins sooner than the 20 second minimum startup time for this system and peaks at a higher flow rate than plant test data indicate and RELAP5YA calculates. HPCI injection terminates at about the same time for the Simulator and RELAP5YA. The wide solid line slightly above the time axis during HPCI injection is steam flow to the HPCI turbine calculated by RELAP5YA; the Simulator's curve for this parameter was not obtained on the data tape for comparison.

Figure 4-2.13 shows the RELAP5YA LPCI mass flow rate history. Although the Simulator was observed to have LPCI injection during the test, values did not appear on the data tape. RELAP5YA calculates that LPCI injection surges into the A loop, beginning at 25 seconds after the low pressure permissive is reached. The injection rate then decreases between 30 and 68 seconds due to pressure buildup in the pipe caused by lower quality fluid exiting the break and wall heat transfer. The injection rate then recovers, and injects fluid into the broken loop after the corresponding LPCI injection valve closes. No injection occurs in the intact loop, since the accident assumes the LPCI-B Injection Valve failed to open on demand.

Figure 4-2.14 compares the LPCS mass flow rate histories. LPCS injection begins at nearly the same time in the Simulator and RELAP5YA computations. RELAP5YA calculates that the two LPCS systems together inject a total of 1,200 lbm/sec, whereas the Simulator injects only 1,000 lbm/sec. This latter value is conservatively low relative to plant test data.

Figure 4-2.15 shows core mid-plane clad temperatures for the three regions of the RELAP5YA analysis. Very little heatup can be observed. The Simulator models the core as only one average fuel region divided into 24 axial nodes. Figure 4-2.16 shows average bundle clad temperatures for several axial locations of the Simulator and RELAP5YA. The Simulator shows clad heatup to as high at 1,145°F while RELAP5YA only heats to 590°F. The differences in clad heatup are due to differences in void fractions calculated by the two models.

Figure 4-2.17 shows bundle void fraction histories. The core voiding and recovery sequence can be seen for the Simulator and RELAP5YA. RELAP5YA recovers at 94 seconds while the Simulator does not completely recover until after 200 seconds.

The containment drywell and wetwell pressure histories from the Simulator and RELAP5YA calculations are compared in Figure 4-2.18. A comparison of respective parameters shows similar trends although the magnitudes differ somewhat during the transient. The Simulator parameters reach asymptotic values that are within 5 psi of the RELAP5YA results. These results are acceptable since these parameters are displayed on a back panel in the control room and would only be checked periodically by the operator.

Figure 4-2.19 shows containment gas temperature histories. The temperatures calculated by the Simulator are higher in both drywell and wetwell than those calculated by RELAP5YA.

Figure 4-2.20 shows the torus liquid temperature histories. The Simulator and RELAP5YA values remain within a few degrees of each other.

In summary, the Simulator shows good agreement with RELAP5YA calculations. The differences are mainly for parameters where the coarseness of Simulator modeling does not permit any greater accuracy, such as its modeling of only one recirculation loop and in its lumping of the bypass and core regions.

4.3 SMALL BREAK LOCA

The small break LOCA (SB-LOCA) case is a single break in the suction pipe of one recirculation loop. The Simulator break size was 0.0615 ft^2. The RELAP5YA break size was slightly smaller, 0.05 ft^2. The initial conditions for the SB-LOCA for both the Simulator and RELAP5YA are the same as for the LB-LOCA and are listed in Table 4.21.t.1.

4.3.1 SB-LOCA Accident Assumptions

The accident assumptions are summarized in Table 4-3.1. The major assumptions include loss of auxiliary power, failure of RCIC to auto start, failure of HPCI to auto start, and no CRD flow. Thus, available systems for injecting water to the reactor vessel are:

a 2 Low Pressure Core Spray systems (LPCS) taking suction from the torus

b 2 Low Pressure Core Injection systems (LPCI) taking suction from the torus

c The Automatic Depressurization System (ADS) may activate if appropriate signals exist.

The nodalization for the VY-NSSS and containment are similar to the large break model and are shown in Appendix A.2. The integral model contains 170 volumes, 184 junctions, and 176 heat structures.

4.3.2 SB-LOCA Comparison

Table 4-3.2 summarizes the timing of significant events for this case. The Simulator and RELAP5YA calculations are compared in

Figures 4-3.1 through 4-3.20.

Figure 4-3.1 compares the Simulator and RELAP5YA power histories. The Simulator shows a more rapid shutdown than RELAP5YA over the first three second. By six seconds, both reach the same value. The Simulator has a slightly greater break area, hence, greater break flow and voiding, which could account for part of the difference in early power history.

Figure 4-3.2 shows steam dome pressure histories. After an initial decline, the pressure rises due to MSIV closure. Following, ADS at 138 seconds for RELAP5YA and 232 seconds for the Simulator, the pressure declines sharply. The pressure continues a slow decline until vessel flooding causes repressurization. The pressure histories are the same for the Simulator and RELAP5YA up until the time of MSIV closure. RELAP5YA pressure history diverges from the Simulator at ADS, because its LO-LO level signal occurred about 80 seconds earlier than the Simulator. After ADS, the Simulator shows a steeper pressure decline than the more gradual decline of RELAP5YA. Also, the Simulator steam dome pressure rises to about 80 psia following vessel flooding at 450 seconds. The RELAP5YA pressure rises to 180 psia following vessel flooding at 525 seconds.

Figure 4-3.3 shows narrow-range level histories. The Simulator shows a wider range of values than RELAP5YA. The RELAP5YA values are limited to the displayed values in the range of 77" to 177". The Simulator and RELAP5YA level histories are fairly similar. The Simulator has a more gradual decline than RELAP5YA up to 96 seconds, when LO-LO level is reached, which is about 80 seconds after RELAP5YA. The Simulator shows a slight level swell following ADS. Later, at 350 seconds, the Simulator shows a recovery from its minimum value. RELAP5YA shows a fairly sustained recovery about 100 seconds later than the Simulator. The early recovery is due to early injection of LPCI and LPCS in the Simulator calculation.

Figure 4-3.4 shows wide-range level histories. Three RELAP5YA histories and a single Simulator history are shown. Plant instrumentation will respond to the maximum of the inner and outer shroud values. The Simulator history follows the same trend as the RELAP5YA outer shroud level, but it does not uncover as deeply as RELAP5YA. Full recovery (177") is achieved by around 400 seconds for both the Simulator and RELAP5YA. The Simulator displayed value would be limited to 177". The RELAP5YA calculated results are limited to the displayed values.

Figure 4-3.5 shows feedwater and steam line flow histories. The Simulator and RELAP5YA feedwater flow histories are similar. The RELAP5YA calculation ramped the flow to zero in 5 seconds. The Simulator feedwater response was nearly the same, and the flow decreased to zero in about 8 seconds.

Steam line flow histories are based on the timing of three events; turbine stop valve closure, bypass valve opening, and MSIV closure. Simulator turbine stop valve closure occurs about 3.5 seconds earlier than for RELAP5YA. The bypass valves are fully open in the RELAP5YA calculation while the Simulator modulates the area change for these valves. This explains the higher steam flows calculated by RELAP5YA. The MSIV closes at about the same time in both calculations.

Figure 4-3.6 shows pump speed histories. Two RELAP5YA histories are shown, one for the Loop A (broken loop) recirculation pump and one for the Loop B (intact loop) pump. The Simulator calculates only one behavior for both pumps. The Loop A and Loop B histories are essentially the same in the RELAP5YA calculation. The Simulator shows a similar behavior. The RELAP5YA recirculation pump's coastdown ends at about 33 seconds while the Simulator calculates that the coastdown ends at about 60 seconds.

Figure 4-3.7 presents the break flow histories. Following MSIV closure, RELAP5YA calculates an increase in break flow rate which matches the increase in pressure. The Simulator does not predict this increase in break flow rate. The major difference in the break flow rate is due to the timing of ADS which occurs earlier in the RELAP5YA calculation due to a more sudden drop in narrow range level. As the vessel refills, the break flow calculated by RELAP5YA increases due to presence of liquid at the break. The Simulator does not mirror this behavior.

Figure 4-3.8 shows recirculation loop flow rate histories. The Simulator shows only an intact loop behavior for both loops. The RELAP5YA histories for both Loop A and B are identical and they are close to the Simulator calculation. As in Figure 4-2.7, the Simulator calculates a slightly slower coastdown after 20 seconds.

Figure 4-3.9 shows jet pump flow histories. RELAP5YA Loops A and B jet pump bank flow rates are almost identical. The Simulator calculation for the jet pump flow rate is consistent with the RELAP5YA calculation; however it calculates slightly higher flow rates for almost the entire transient.

Figure 4-3.10 shows core inlet mass flow rate histories. Core only and core plus bypass values are shown for RELAP5YA. The Simulator does not have a separate bypass region. Core and bypass region are lumped together in the Simulator calculation. Although the Simulator follows the overall trend of the RELAP5YA calculation, it shows slightly higher inlet core flow rates (consistent with Figure 4-3.7) and it does not include the spikes present in RELAP5YA calculation during ECCS injection.

Figure 4-3.11 shows S/RV1 and ADS flow rates histories. The safety relief valve opens only once in the RELAP5YA calculation due to the early opening of the ADS. The flow through the valves was about the same for both codes. At the end of the accident, when the RPV refills with ECCS water, the Simulator should show an increase due to water through the ADS valves as calculated by RELAP5YA.

Figure 4-3.12 shows LPCS mass histories. Due to the faster system depressurization calculated by the Simulator (see Figure 4-3.2) the LPCS commences about 70 seconds earlier than in the RELAP5YA calculation. RELAP5YA LPCS reaches 1200 lbm/sec while the Simulator reaches only 900 lbm/sec due to different input for the pump characteristics. The Simulator uses more conservative data from the VY FSAR while RELAP5YA uses conservative VY plant test data.

Figure 4-3.13 shows the RELAP5YA LPCI mass flow rate history. Although the Simulator has LPCI injection during the test, values did not appear on the data tape. LPCI injection into both loops begins at 337 seconds in RELAP5YA calculation.

Figure 4-3.14 shows core mid-plane clad temperatures for the RELAP5YA analysis. Peak clad temperature of 725°F are reached in the high power bundle at about 343 seconds. The Simulator models the core as only one average fuel region divided into 24 axial nodes. Figure 4-3.15 compares the average bundle clad temperatures for several axial locations calculated by RELAP5YA and the Simulator. RELAP5YA shows clad heatup as high as 690°F at about 340 seconds in the transient while the Simulator clad temperatures follow the saturation temperature at all axial locations.

A better understanding of the differences in clad temperatures calculated by the two codes is provided in Figures 4-3.16 and 4-3.17. The two figures show the average bundle void fraction histories for RELAP5YA and the Simulator, respectively. RELAP5YA calculates more voiding in the core during the period of clad heatup. The Simulator calculates a maximum core void fraction of about 0.85. At this void fraction there is enough liquid in the core to maintain the rods close to saturation temperatures.

The containment drywell and wetwell pressure histories from the Simulator and RELAP5YA calculations are compared in Figure 4-3.18. The comparison of respective parameters shows similar trends although the magnitudes differ. The Simulator pressures in both the drywell and wetwell are about 5 psia higher than RELAP5YA results. This can possibly be attributed to the present RELAP5YA containment nodalization which precludes circulation within the wetwell pool; hence not enough heatup of the wetwell gas space is calculated.

Figure 4-3.19 shows drywell and wetwell containment gas temperature histories. The temperatures calculated by the Simulator are about 30°F higher in the drywell and about 18°F in the wetwell.

Opposite results are obtained for the torus liquid temperature histories, presented in Figure 4-3.20, which show higher pool temperatures calculated by RELAP5YA. The higher liquid temperature calculated by RELAP5YA is attributed to the present containment nodalization which does not allow the hotter water to raise to the pool surface and heatup the gas space.

In summary, the Simulator is able to predict the correct trends for this accident. For many parameters, the Simulator shows adequate agreement with the RELAP5YA calculations.

4.4 MAIN STEAM LINE BREAK IN THE STEAM TUNNEL

The event analyzed is the double-ended guillotine break in one main steam line located in the steam tunnel coincident with loss of off-site power.

4.4.1 MSLB Accident Assumptions

The initial plant operating conditions for this accident for both the Simulator and RELAP5YA are summarized in Table 4-4.1. These parameters reflect VYNPS operation at conservative design power conditions (e.g., 105% of reactor vessel rated steam flow, severe core radial and axial power profiles) rather than rated conditions.

The assumptions used for this accident are summarized in Table 4-4.2. In addition to the assumed steam line break and loss of off-site power, severe conditions are imposed by the assumed failure of HPCI, one LPCS and two LPCI Systems to inject emergency coolant upon demand.

4.4.2 MSLB Comparison

Table 4-4.3 summarizes the timing of significant events for this accident for both RELAP5YA and the Simulator. This table is provided as aid to review the following figures that contain results.

Figures 4-4.1 through 4-4.19 depict plant response for this accident calculated by the RELAP5YA code and the Simulator.

Figure 4-4.1 is a plot of Core Thermal Power. The RELAP5YA predicted core power history is in good agreement with the Simulator calculated power. Both reach the same decay heat values beyond 5 seconds in the transient.

Figure 4-4.2 shows the steam dome pressure history. The pressure initially decreases due to the steam line break until the MSIVs are closed at 10.5 seconds. Then the pressure increases due to the addition of core decay heat to the coolant until the safety relief valve (S/RV1) setpoint of 1080 psig is reached at 36.0 seconds for RELAP5YA and about 120 seconds for the Simulator. RELAP5YA predicts more frequent cycling of the safety relief valve (S/RV) because the S/RV's reset pressure is 1062.3 psia for RELAP5YA and only 1040 psia for the Simulator. Following manual opening of the Automatic Depressurization System (ADS) at 600 seconds, both codes calculate rapid depressurization of the NSSS; however, the Simulator calculates a faster depressurization. We believe that this difference occurs because the Simulator model removes pure steam with a quality of 1.0 through the ADS valves, where as, RELAP5YA removes wet steam with a quality of 0.5 to 1.0. (See discussion of Figure 4-4.12 for further information). At 1200 seconds both models calculate that the system depressurized to about 72 psia.

Figure 4-4.3 is a plot of Reactor Pressure Vessel Narrow Range Level. Both codes predict an initial rapid level drop due to the steam flowing out the break and due to the collapsing of voids in the core following scram. The level drop is arrested following MSIV closure at about 11 seconds. Following isolation, the Simulator predicts a general increase in core voids to about 0.4 which cause displacement of the water from the core region into the shroud. As a consequence, the Simulator calculates a higher increase in the narrow range level. RELAP5YA calculates less voiding of the core prior to ADS, hence much lower narrow range levels. At 240 seconds RELAP5YA indicated level goes off-scale. After ADS at 600 seconds, RELAP5YA calculated level recovers for a short period of time due to level swell then it goes again off scale. The Simulator does not distinguish between a narrow and a wide range level, i.e., the same calculated parameter is displayed for both indications.

Figure 4-4.4 presents the Reactor Vessel Wide Range Water

Level. The level parameter calculated by the Simulator covers a downcomer level span which is similar to the wide range water level in the outer shroud. As can be seen, the Simulator calculated level in Figures 4-4.3 and 4-4.4 is identical. In contrast to the RELAP5YA calculation, during the period of S/RV's cycling, the Simulator does not calculate a significant level decrease. Both codes calculate the level decrease following ADS, followed by an increase in level after LPCS injection. The early and rapid increase in level, predicted by the Simulator after 700 seconds, is due to the more rapid depressurization (Figure 4-4.2) which causes the LPCS to actuate too soon, compared to RELAP5YA.

Figure 4-4.5 presents the Feedwater Flow Rate histories. The end of coastdown for the Simulator calculation, 8 seconds, compares well with the RELAP5YA value of 5 seconds. The Simulator prediction for this parameter is good with respect to the time scale in which the operators will respond to this accident.

Figure 4-4.6 presents the Main Steam Line Flow Rate histories. The average value of the steam flow rate calculated by the Simulator compares well with the RELAP5YA flow rate through the broken steam line. However, the ringing predicted by the Simulator is believed to be incorrect.

Figure 4-4.7 presents the Steam Line Break Flow Rates. The downstream break flow rate is fed by backflow from the three intact steam lines through the pressure averaging manifold pipes. The MSIV closes at 10.5 seconds. Flow through the downstream break continues for another 15 seconds being fed by the large volume of steam present in the pipes. The upstream break flow is limited by the flow limiter and goes to zero shortly after MSIV closure. We did not obtain the Simulator results for these parameters.

Figure 4-4.8 is a plot of the Recirculation Coolant Pump (RCP) speed. RELAP5YA calculated the two recirculation coolant pump speed histories to be essentially identical. During the first 17 seconds, the pumps coastdown due to decreasing electrical power from the RCP MG sets that are responding to the loss of off-site power. At about 17 seconds the field breaker opens on receipt of a LO-LO level signal and the pumps go into a freewheeling coastdown to about 0.0 rpm by 33 seconds. The values calculated by the Simulator for the RCP's speed compare reasonably well to RELAP5YA. We do not know what produced the inflection in the Simulator curve at 2 seconds.

Figure 4-4.9 presents the Recirculation Loop Flow Rate histories. The Simulator coastdown flow rates are higher than the ones calculated by RELAP5YA, indicating further improvements are needed in the Simulator recirculation loop models.

Figure 4-4.10 presents the Jet Pump Bank Flow Rates histories. The initial jet pump coastdown up to the time of MSIV closure at 10.5 seconds is well predicted by the Simulator. The RELAP5YA flow rates rapidly coastdown and oscillate about zero values as the S/RV cycles. In contrast, the Simulator predicts the jet pump flow to coastdown but remains at more positive values during the remainder of the accident.

Figure 4-4.11 is a plot of the Core Inlet Flow. For the first 600 seconds, the core inlet flow for both RELAP5YA and the Simulator follows the respective jet pump behavior. After manual ADS, both codes indicate positive core flow, primarily due to lower plenum flashing which subsides after 690 seconds.

Figure 4-4.12 shows the flow rates for the safety/relief valve with the lowest setpoint (S/RV1) and the Automatic Depressurization System (ADS) that consists of all four safety/relief valves. For the first 600 seconds of the transient, the flow rates through the S/RV predicted by the Simulator and RELAP5YA are essentially identical. Following ADS, the Simulator ADS flow rate decreases more rapidly than RELAP5YA because the system pressure decreases more rapidly (Figure 4-4.2). Based on the RELAP5YA code assessment work against Two Loop Test Apparatus (TLTA) Small Break 6432

(Reference 4.4.1) and FSTF Mark I Containment Test (Reference 4.4.2), we believe RELAP5YA is providing a realistic prediction of the RPV pressure and ADS mass flow rates.

Figure 4-4.13 shows the LPCS Mass Flow Rate. The maximum flow rate for the LPCS pump in the Simulator calculation is lower than the one predicted by RELAP5YA. The Simulator's ECCS characteristics reflect very conservative values from VY FSAR whereas RELAP5YA uses slightly conservative values from VY Plant test data. The LPCS injection in the Simulator calculation starts early because it reaches the low pressure permissive early.

Figure 4-4.14 compares the Average Bundle Peak Cladding Temperature. Both codes calculate that the fuel rods are well cooled by nucleate boiling through out the first 800 seconds of the accident. At about 800 seconds, the RELAP5YA calculates that the cladding begins to heatup due to the transition to film boiling caused by the relatively high local void fractions (see Figure 4-4.15) and low flow rates. This heatup period is arrested at about 900 seconds due to the injection of LPCS that begins to enter the core. The cladding at all core location is finally well cooled by 1050 seconds. The Simulator does not calculate any rod heatup for this accident due to early actuation of LPCS.

Figure 4-4.15 shows the Core Average Void Fraction history calculated by RELAP5YA. After the MSIVs close at 10.5 seconds, the void fraction at 81 inch elevation stabilizes at a periodic value of 20%. The void fraction oscillations up to 600 seconds are caused by the cycling of S/RV1. The rapid depressurization, due to ADS actuation at 600 seconds, causes the void fraction to rapidly increase to about 60%. The core voiding rate increases between 600 and 863 seconds due to flashing and boiling. Shortly after LPCS injection is initiated at 863 seconds, the core average void fraction decreases. This results from LPCS coolant entering the core region from the top down through some upper tie plates, and from the relatively large bypass region through the drilled holes in the fuel support pieces up into the fuel assemblies. Note that the core average void fraction is above 95% between 800 seconds and 1000 seconds. Figure 4-4.14 shows that this corresponds to the core heatup period.

Figure 4-4.16 shows the Average Core Void Fraction history calculated by the Simulator. After MSIV closure, the void fraction stabilizes at a value of 40%, compared to 20% in Figure 4-4.15. Following ADS, the core void fraction increases. However, due to early LPCS injection the core liquid inventory depletion was not as severe as in the RELAP5YA calculation, resulting in the prediction of no heatup.

The Containment Drywell and Wetwell Pressure histories from the Simulator and RELAP5YA calculations are compared in Figure 4-4.17. The comparison of respective parameters shows similar trends although the magnitude differ somewhat during the transient. The drywell pressure in the Simulator calculation increases slowly due to the loss of the drywell coolers on the loss of offsite power. During normal operation, the drywell coolers remove heat convected from the NSSS to the drywell gas environment. The RELAP5YA calculates a sudden dip in pressure at 600 seconds. We believe that a better nodalization in the RELAP5YA containment model such as the one used in Reference 4.4-2 will improve the RELAP5YA predictions. The wetwell pressure calculated by RELAP5YA is about 1.2 psia higher than the one calculated by the Simulator.

Figure 4-4.18 shows Containment Gas Temperature histories. The differences between the drywell temperatures calculated by the two codes are due to different initial conditions. The dip in temperature in the RELAP5YA prediction can be removed with a better nodalization. The wetwell gas temperatures for both codes are similar. For the first 600 seconds, the RELAP5YA calculation shows no increase in the gas space temperature because the present wetwell nodalization does not allow natural circulation within the gas space.

Figure 4-4.19 shows Torus Liquid Temperature histories. The higher liquid temperature calculated by RELAP5YA is due in part to the due to lack of circulation within the pool, which results in a low rate of heat transfer to the gas space.

In summary, the Simulator is able to predict the correct trends for this accident. For many parameters, the Simulator shows adequate agreement with RELAP5YA calculations. The differences are due mainly to the more rapid depressurization after ADS actuation in the Simulator prediction.

TABLE 4-2.1

VY LOCA Benchmark Initial Conditions

Parameter	Simulator	RELAP5YA
Power	100% BOC 1,593 MWt	100% EOC 1,593 MWt
Steam Dome Pressure	1,025 psia	1,020 psia
Core Flow	96% 12,800 lbm/sec	100% 13,300 lbm/sec
Level (above lower reactor vessel invert)	512.3"	512"
Drywell Pressure	16.5 psia	16.6 psia
Drywell Temperature	135.5°F	165.0°F
Wetwell Pressure	14.67 psia	14.7 psia
Wetwell Gas Temperature	79°F	74°F
Wetwell Liquid Temperature	72°F	74°F
Recirculation Pump Speed	1,670 rpm	1,670 rpm

TABLE 4-2.2

Summary of Large Break LOCA Assumptions

1. DEG recirculation suction break ($2*3.64\ ft^2$) occurs at 4.0E-6 seconds in the drywell.

2. Loss of auxiliary power occurs at 4.0E-6 seconds.

3. Reactor SCRAMs after 0.5 second delay from first RPS signal. SCRAM Curve 67B-EOC is used.

4. Feedwater coasts down to 0.0 lbm/sec at 5.0 seconds.

5. MSIVs close in 4.0 seconds after isolation signal plus 0.5 second delay.

6. Recirculation pumps in A and B loops coast down with decreasing power from loss of M/G sets.

7. ADS may actuate if appropriate signals exist. Thereafter, ADS cycles open/close at 12 psid between steam line and drywell when ADS criteria are currently met at any time.

8. HPCI injects upon demand, and terminates on high RPV level or low steam line pressure (<90 psia).

9. RCIC steam turbine valve fails to open.

10. Two LPCS Systems inject upon demand.

11. LPCI A System injects upon demand.

12. LPCI B injection valve fails to open upon demand (single failure).

13. Drywell pressure and temperature are derived from containment model.

14. Wetwell pressure and temperature are derived from containment model.

15. Best Estimate point reactor kinetics parameters for a core initially operating at 1,593 MWth.

16. Best Estimate core heat transfer.

17. Passive heat structures are included

TABLE 4-2.3

Sequence of Events for Large Break LOCA

	Event	Time (seconds) RELAP5YA	Simulator
1.	Break opens	4.0E-6	4.0E-6
2.	Loss of normal auxiliary power	4.0E-6	4.0E-6
3.	High drywell pressure (p >2.5 psig)	0.035	
4.	Turbine stop valves starts to close (and are completely closed in 0.1 second)	0.044	
5.	Reactor SCRAM on high drywell pressure	0.5	
6.	Initiate turbine bypass valve opening	0.5	
7.	Turbine bypass valve completely open	1.1	
8.	Low level signal (127 inches) occurs	0.7	
9.	RPS MG set underfrequency (57 Hz) condition occurs.	3.0	
10.	MSIVs begin to close on RPS underfrequency	3.5	
11.	Low low level signal (82.5 inches) occurs	3.8	
12.	Control rods are fully inserted	4.2	
13.	Feedwater flow coasts down to zero	5.0	18.0
14.	Lower plenum flashing begins	5.0	
15.	Earliest nodal heatup	16	8
16.	MSIVs are completely closed	8	10
17.	Recirculation pumps trip on LO-LO level plus 10.3 second delay	14.1	
18.	HPCI injection begins (high DW pressure + 20.3 seconds)	20	11
19.	Recirculation Loop B (intact) discharge valve begins to close	24	
20.	LPCS injection begins	32	30
21.	Minimum Primary System inventory (82,000 lb) occurs	49	
22.	Peak clad temperature occurs (590°F [1145°F])	50	188
23.	Recirculation Loop B discharge valve closed	57	
24.	HPCI flow terminated on low pressure signal	40	48
25.	Core is well cooled	80	218

TABLE 4-3.1

Summary of Small Break LOCA Assumptions

1. Small recirculation suction break (0.05 ft^2) occurs at 4.0E-6 seconds in the drywell.

2. Loss of auxiliary power occurs at 4.0E-6 seconds.

3. Reactor SCRAM curve 67B-EOC is used.

4. Feedwater coasts down to 0.0 lbm/sec at 5.0 seconds.

5. MSIVs close in ten seconds after isolation signal plus 0.5 second delay.

6. Recirculation pumps in Loops A and B coast down with decreasing power from loss of MG sets.

7. ADS may actuate if appropriate signals exist. Thereafter, ADS cycles open/close at 12 psid between steam line and drywell when ADS criteria are currently met at any time.

8. HPCI fails to inject upon demand (single failure).

9. RCIC steam turbine valve fails to open.

10. Two LPCS Systems inject on demand.

11. LPCI A and B Systems inject upon demand.

12. Drywell pressure and temperature are derived from containment model.

13. Wetwell pressure and temperature are derived from containment model.

14. Best Estimate point reactor kinetics parameters for a core initially at 1,593 MWth.

15. Best Estimate core heat transfer.

16. Passive heat structures are included.

TABLE 4-3.2

Sequence of Events for Small Break LOCA

	Event	Time (seconds) RELAP5YA	Simulator
1.	Break Opens.	4.0E-6	4.0E-06
2.	Loss of normal auxiliary power.	4.0E-6	4.0E-06
3.	Control rod insertion initiated 0.5 second beyond estimated RPS underfrequency reactor trip signal.	3.5	
4.	MSIVs begin to close.	3.5	
5.	Feedwater flow coasts down to zero.	5.0	7.5
6.	High drywell pressure.	3.5	
7.	Turbine stop valve begins to close.	3.9	
8.	Turbine bypass valve begins to open.	4.0	
9.	Turbine bypass valve completely open.	4.6	
10.	MSIVs completely closed.	13.6	17.5
11.	LO-LO level signal.	17.3	96
12.	Recirculation pump motors trip on low frequency at their MG sets.	17.0	
13.	ADS valves open.	138	232
14.	Earliest nodal CHF.	224	
15.	LPCS injection begins.	332	263
16.	Recirculation loop discharge valves begin to close.	332	
17.	Minimum primary system inventory (161,927 lb) occurs.	342	
18.	Peak clad temperature occurs (692/625)	340	232
19.	LPCI begins to inject.	337	
20.	Core is well cooled.	363	279

TABLE 4.4.1

VY Main Steam Line Break Initial Conditions

Parameter	Simulator	RELAP5YA
Power	104.5% IC41 1,664 MWt	104.5% EOC 1,664 MWt
Steam Dome Pressure	1,032 psia	1,032 psia
Core Flow	96% 13,380 lbm/sec	100% 13,300 lbm/sec
Level (above lower reactor vessel invert)	510.3"	510.5"
Drywell Pressure	16.5 psia	16.6 psia
Drywell Temperature	135.5°F	165.0°F
Wetwell Pressure	14.67 psia	14.7 psia
Wetwell Gas Temperature	79°F	74°F
Wetwell Liquid Temperature	72°F	74°F
Recirculation Pump Speed	1,670 rpm	1,670 rpm

TABLE 4-1.2

Summary of Main Steam Line Break Assumptions

1. A double-ended guillotine break occurs at 4.0E-06 seconds in one main steam line located 25 feet outside the drywell.

2. A coincident loss of offsite power occurs at 4.0E-6 seconds.

3. Feedwater flow coasts down to zero by 4 seconds.

4. Reactor SCRAM occurs after a 0.5 second delay from the first RPS signal. SCRAM curve 67B-EOC is used.

5. MSIVs close in 10 seconds after the first isolation signal plus a 0.5 seconds delay.

6. Recirculation pumps in the A and B loops coastdown with decreasing power from the loss of power to the MG sets. The recirculation pump motors trip 10.3 seconds after the receipt of the LO-LO level signal.

7. HPCI and RCIC steam turbine valves fail to open. Thus, HPCI and RCIC fail to inject upon demand.

8. ADS is manually initiated at 600 seconds.

9. Only one of two LPCS systems upon demand.

10. LPCI Loop A and B injection valves fail to open. Thus, no LPCI injection occurs upon demand.

11. Drywell and wetwell gas temperatures and pressures are relatively stable.

TABLE 4-4.3

Sequence of Events for Main Steam Line Break Case

	Event	Time (seconds) RELAP5YA	Simulator
1.	Break Opens.	4.0E-6	0.0
2.	Loss of normal auxiliary power.	4.0E-6	0.0
3.	High MSL flow logic activated (after 0.5 second delay	0.5	0.5
4.	MSIVs reach 10% closure	1.5	1.5
5.	RPS SCRAM initiated on MSIV closure (after 0.5 second delay)	2.0	2.0
6.	Turbine trip initiated on RPS SCRAM	2.1	
7.	Turbine Stop Valves closed	2.2	
8.	RPS MG set underfrequency trip (after 0.43 second delay)	3.5	
9.	Feedwater flow coastdown to zero	5.0	8.0
10.	Control rods fully inserted	5.7	~5.
11.	Recirculation pump motor trip on spurious LO-LO level signal (after 10.3 second delay)	10.3	
12.	MSIVs fully closed	10.5	11.0
13.	Diesel Generators supplying power to emergency buses	17.0	
14.	Recirculation pumps coastdown to zero	33.0	50.0
15.	HPCI injection normally available but failed by assumption	38.4	
16.	Core average void decreased to 17% from normal 41%	60.0	120.0
17.	Safety/Relief Valve 1 lifts for the first time (1,080 psig setpoint)	36.0	120.0
18.	Manual ADS initiated	600.0	600.0
19.	Core heatup begins due to high fuel assembly void fraction	800.0	
20.	LPCI injection permissive (315 psi) reached	838.3	~660.0
21.	LPCS pump at rated speed	843.8	
22.	LPCS pump discharge valve opens	846.8	
23.	LPCS injecting at rated capacity	863.0	~699.0
24.	LPCI injection normally available but failed by assumption	862.8	
25.	Maximum core average void fraction	875.0	714.0
26.	PCT of 580°F reached in average core bundle	967.0	
27.	All fuel rods well cooled	985.0	

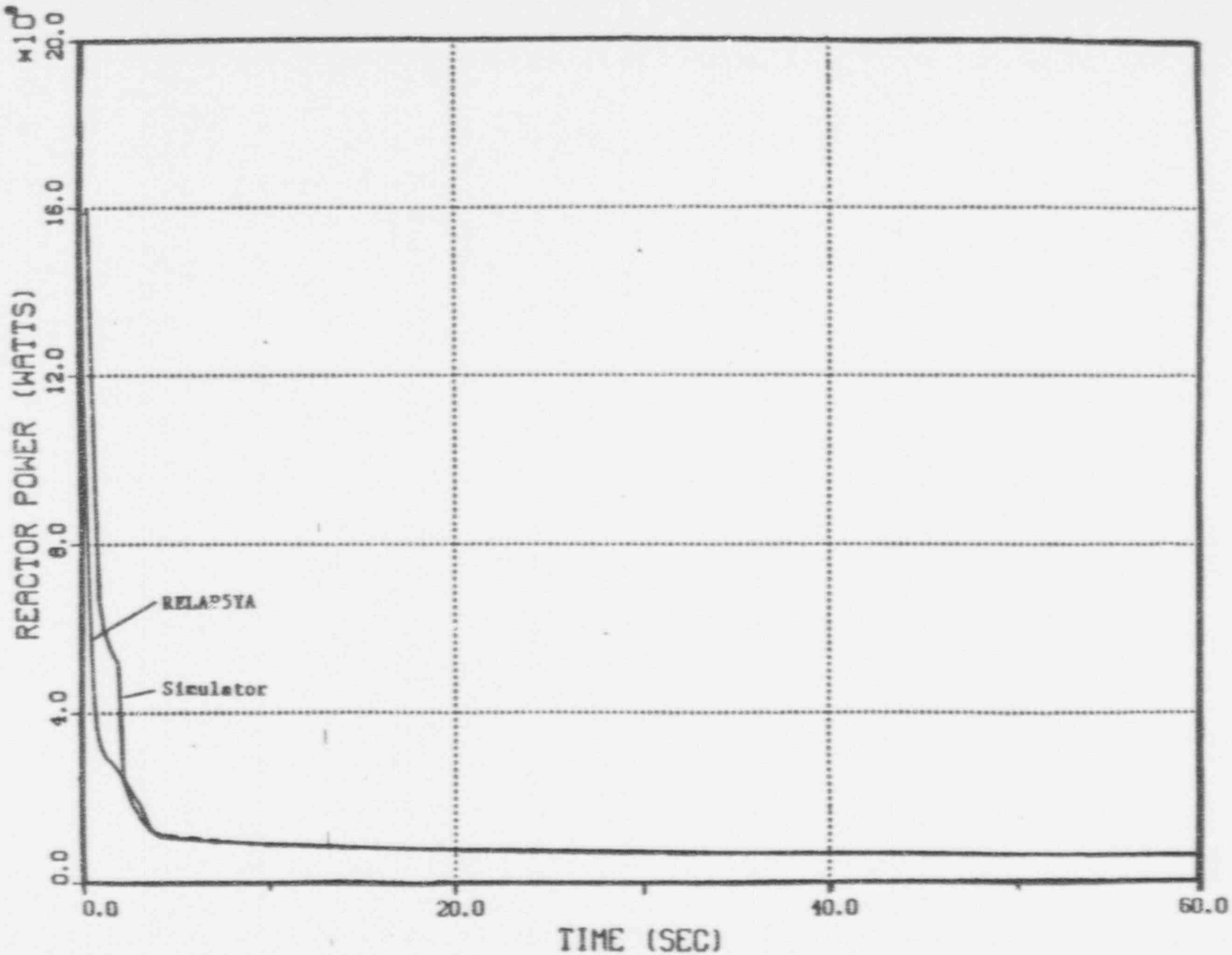


Figure 4-2.1 VY LB-LOCA Core Thermal Power

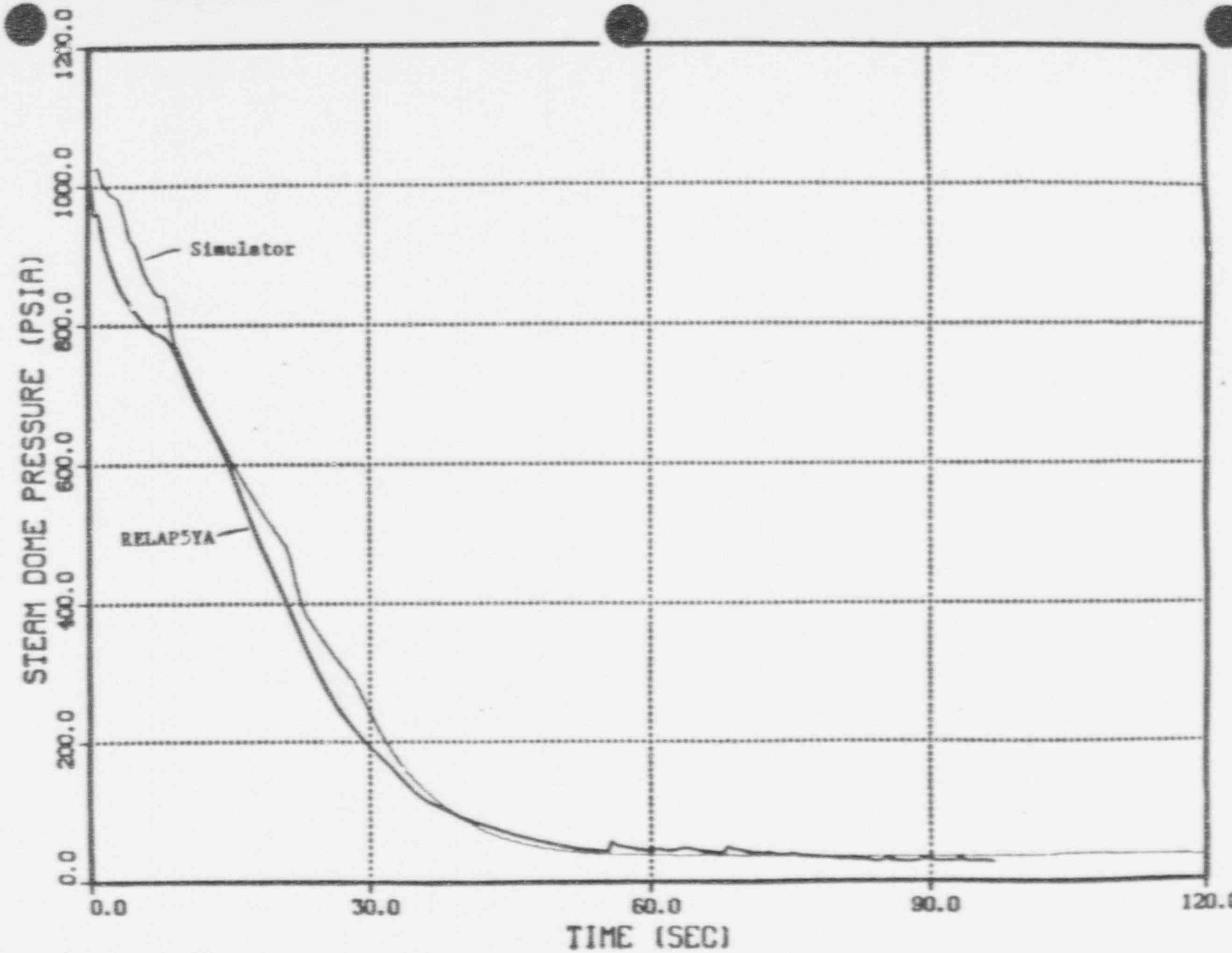


Figure 4-2.2 VY LB-LOCA Reactor Vessel Steam Dome Pressure

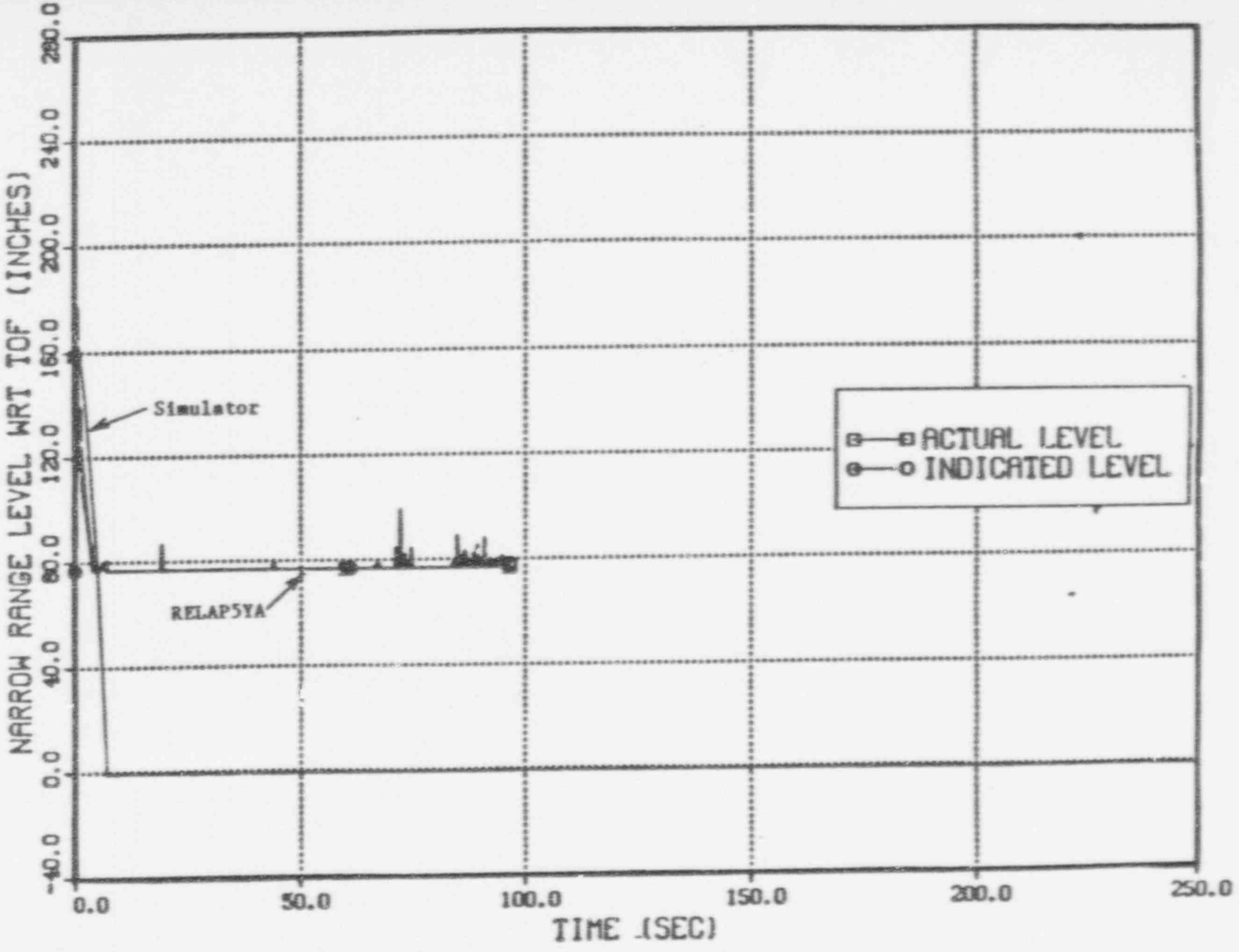


Figure 4-2.3 VT LB-LOCA Reactor Vessel Narrow Range Water Level

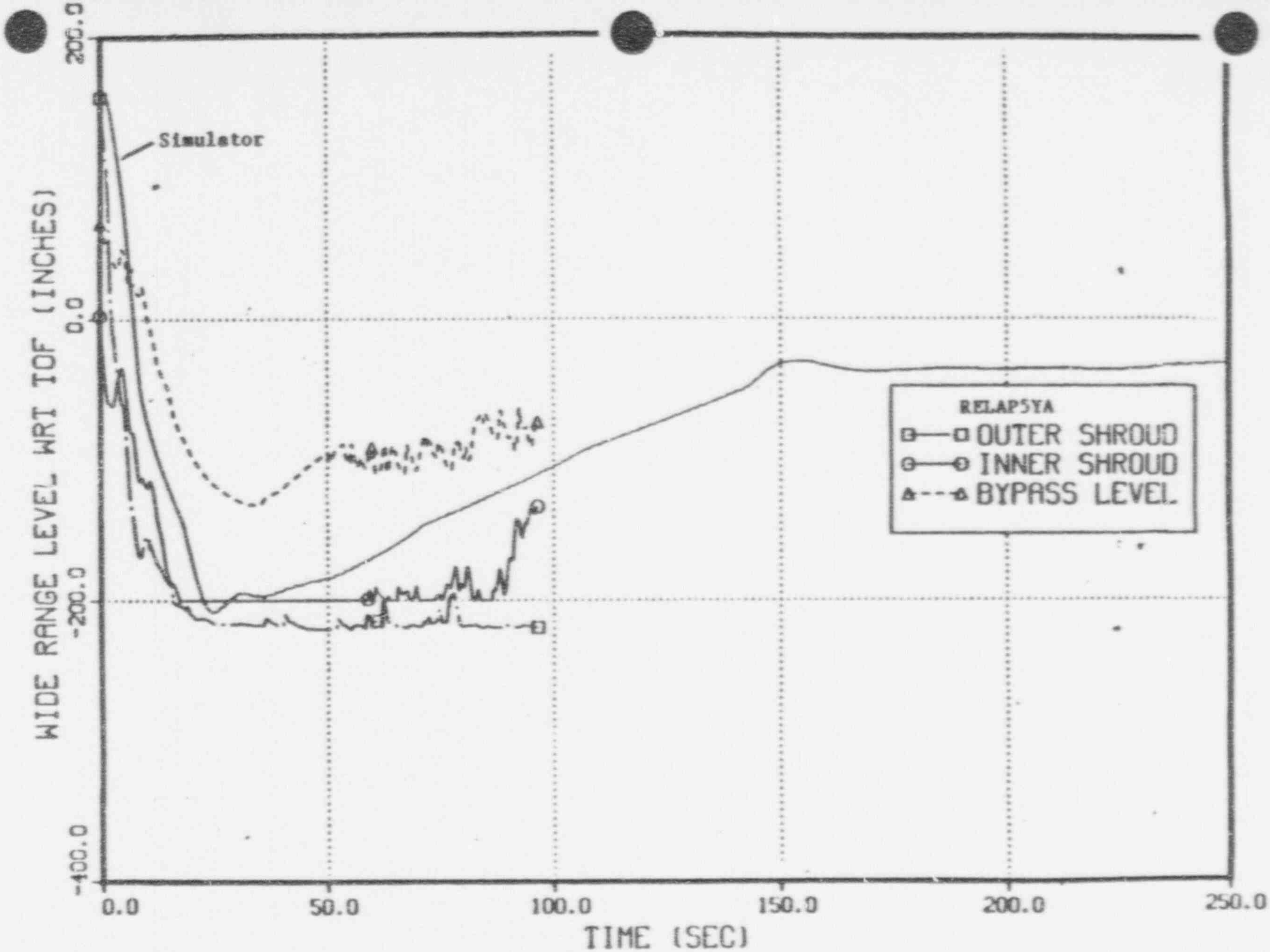


Figure 4-2.4 VT LB-LOCA Reactor Vessel Wide Range Water Level

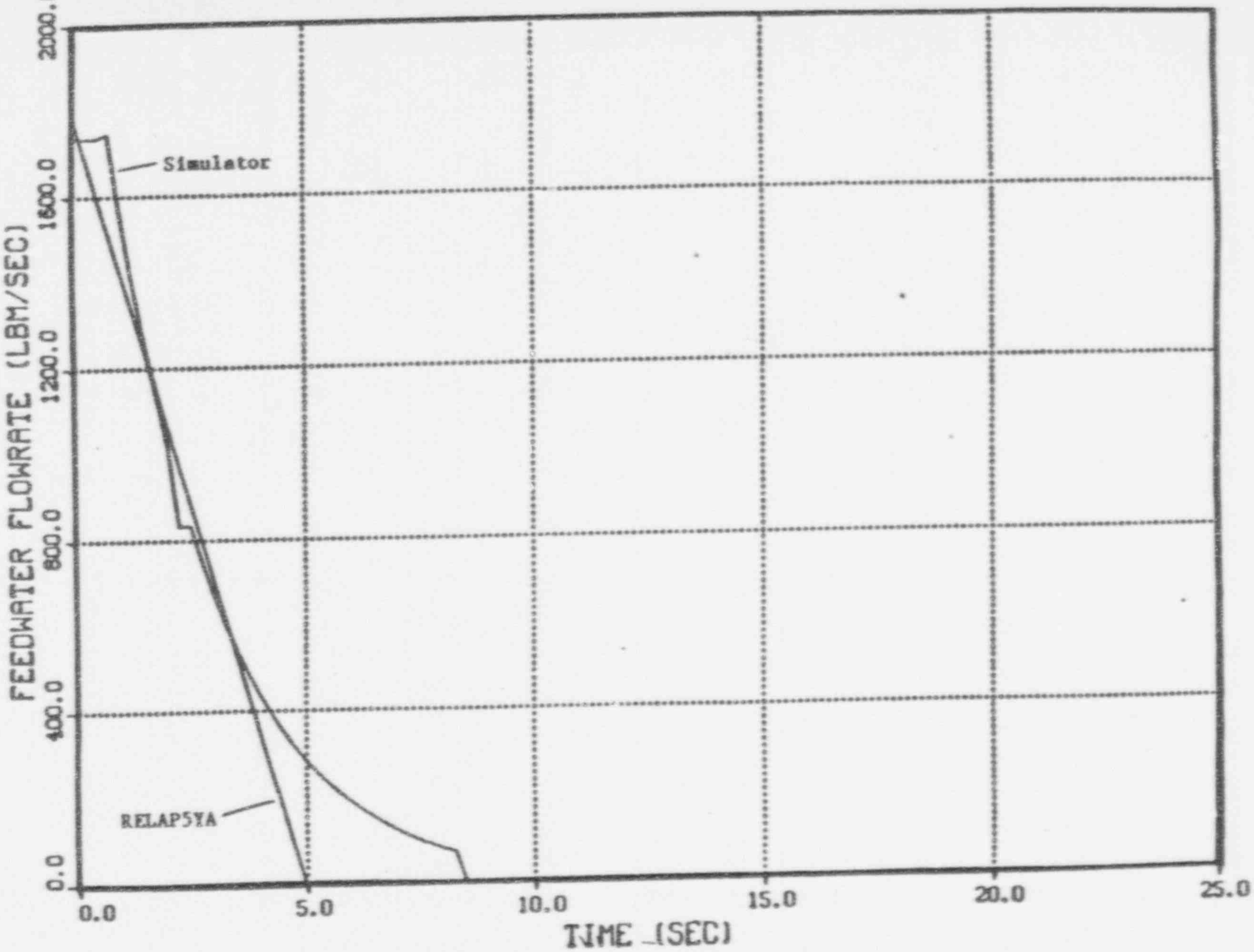


Figure 4-2.5 VY LB-LOCA Feedwater Flow

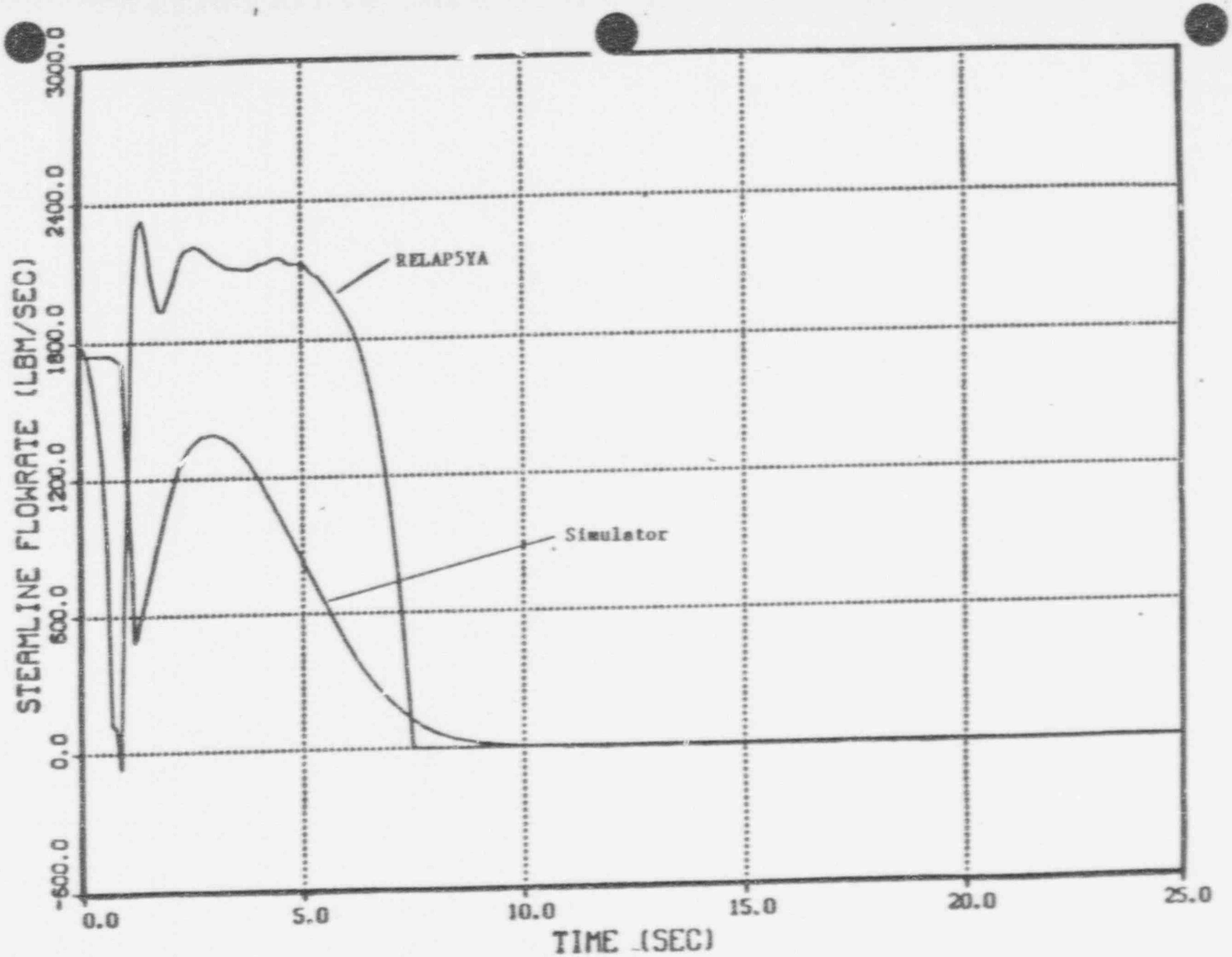


Figure 4-2.6 VT LB-LOCA Steam Line Flow

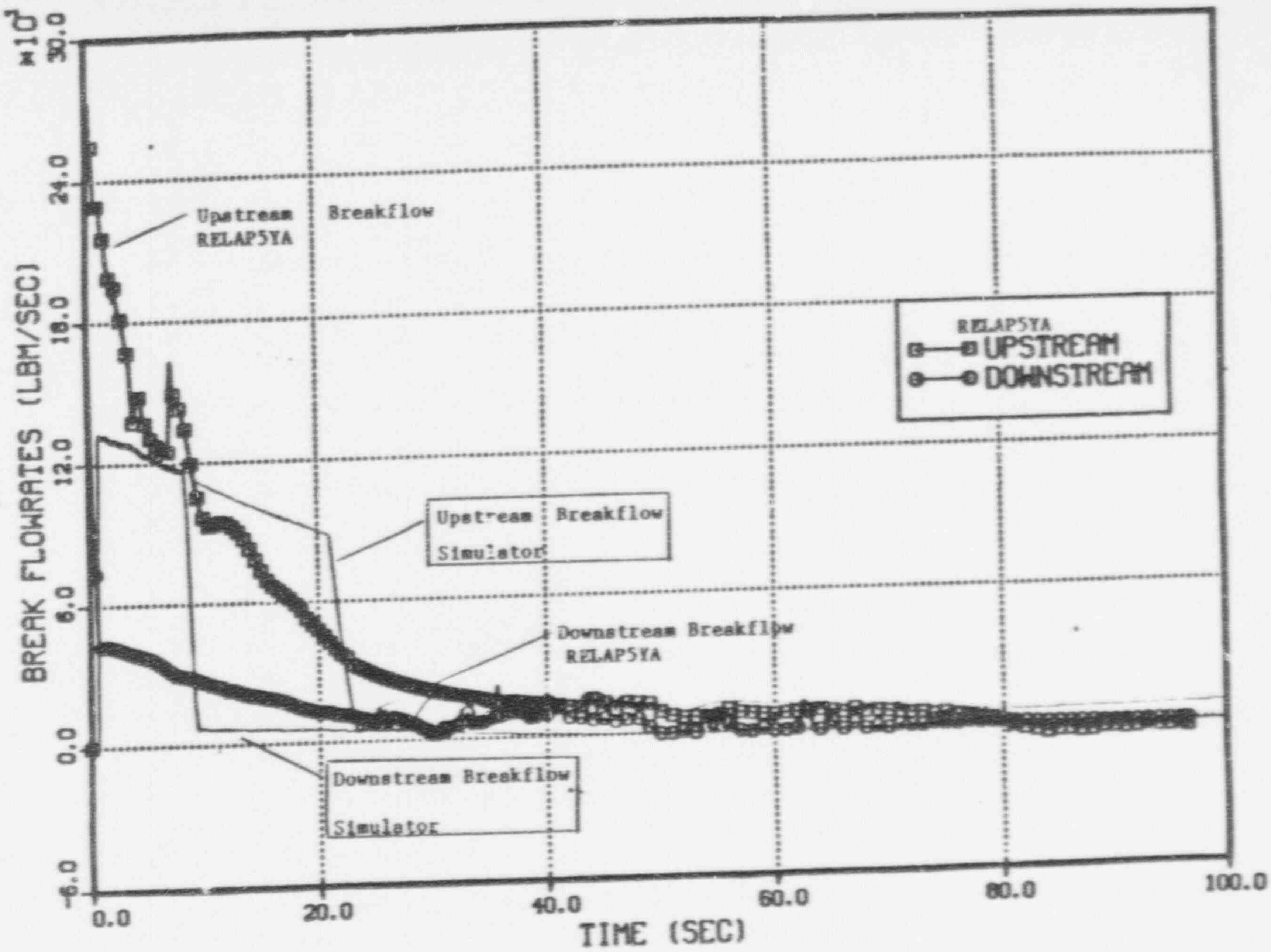


Figure 4-2.7 VY LB-LOCA Break Flows

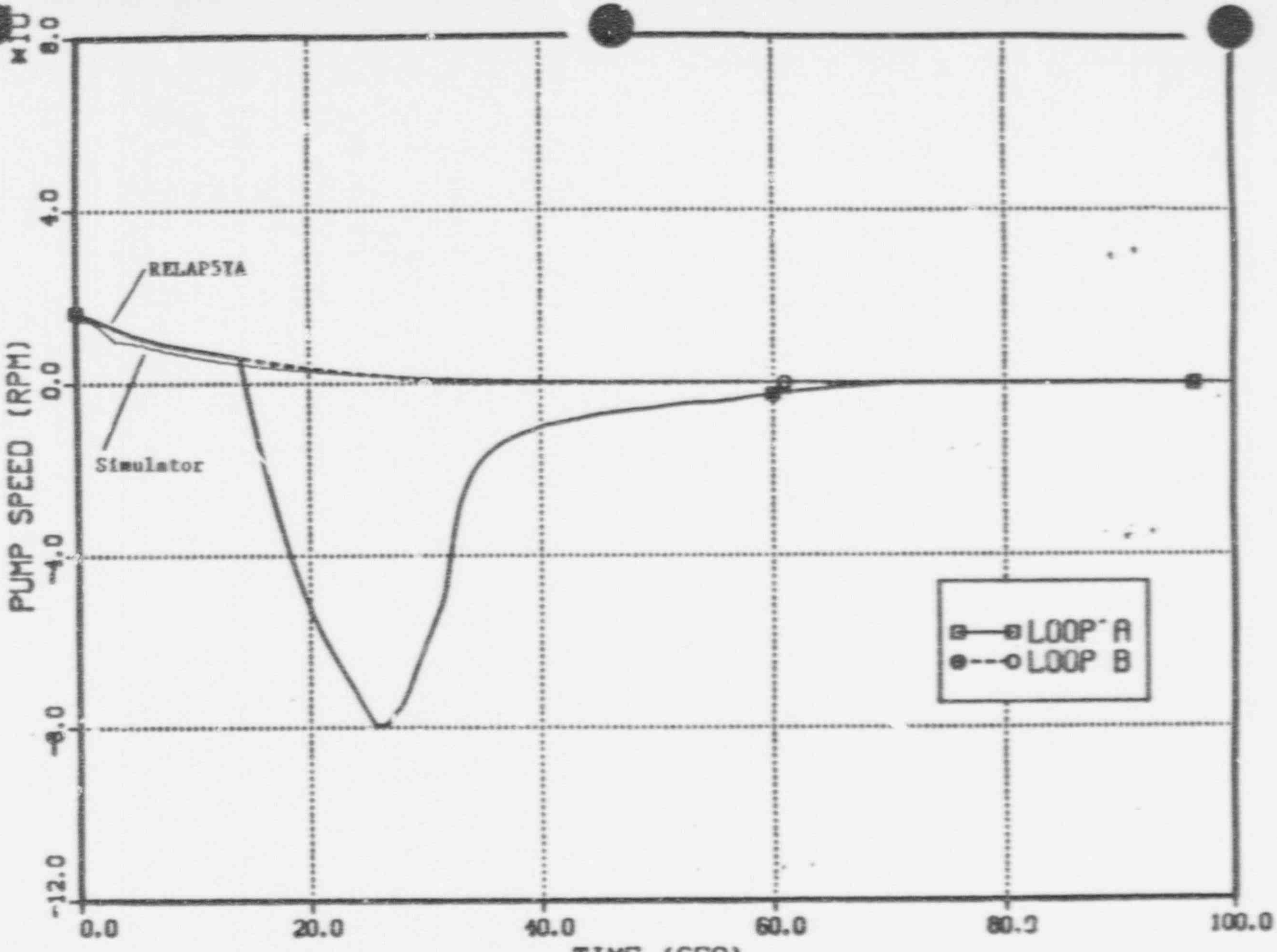


Figure 4-2.8 VY VY LB-LOCA Reactor Recirculation Loop Pump Speeds

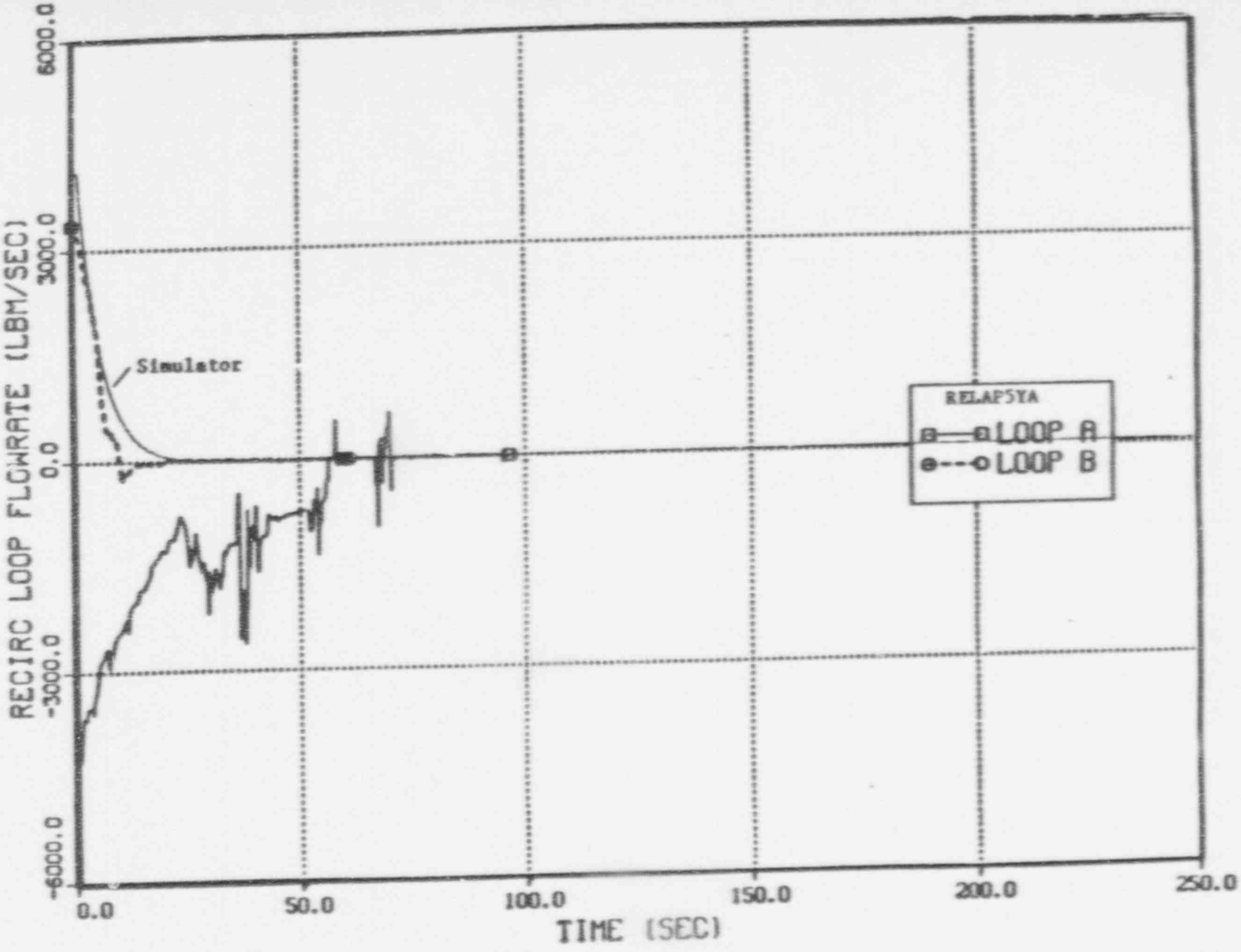


Figure 4-2.9 VY LB-LOCA Reactor Recirculation Loop Flows

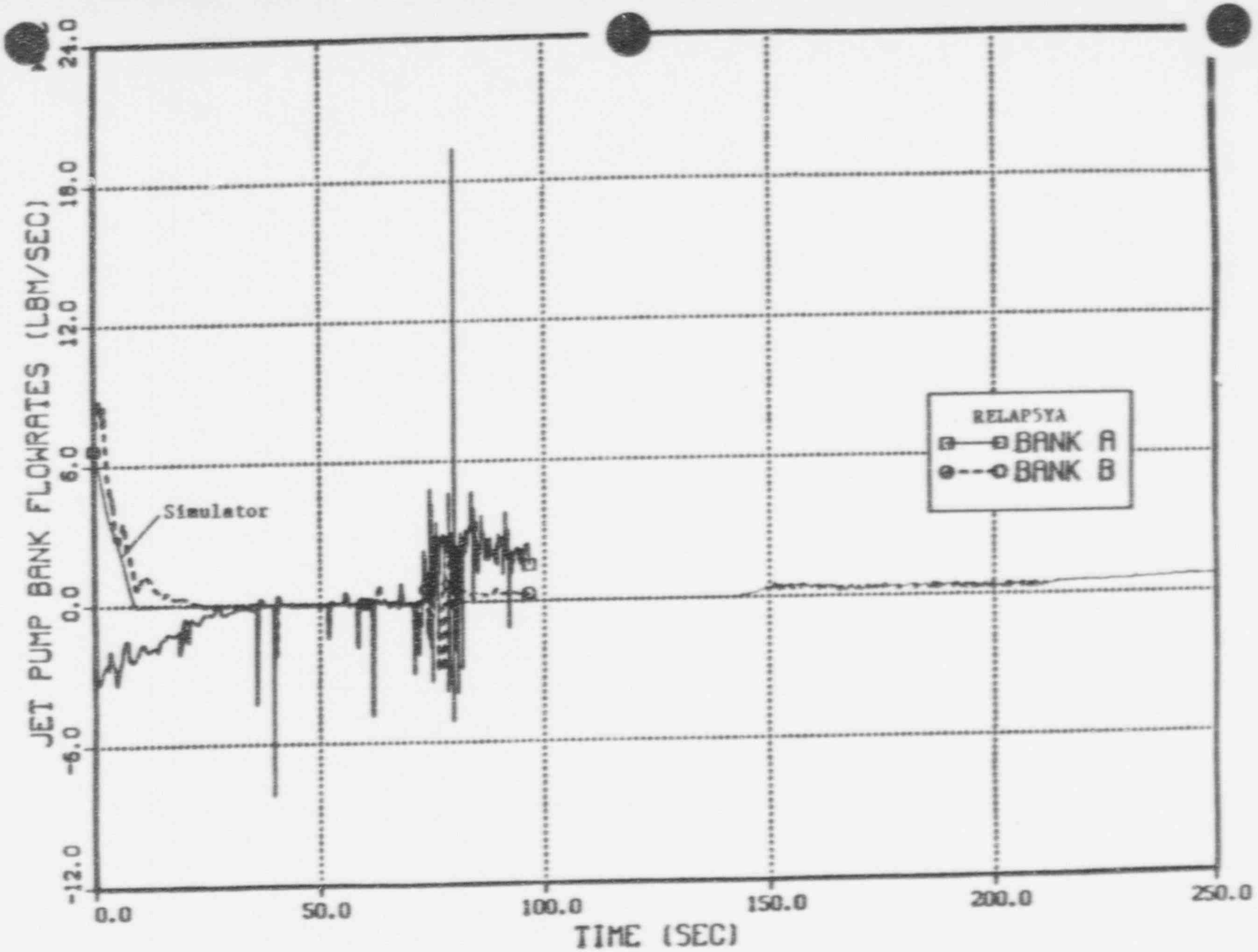


Figure 4-2.10 VY LB-LOCA Jet Pump Flows

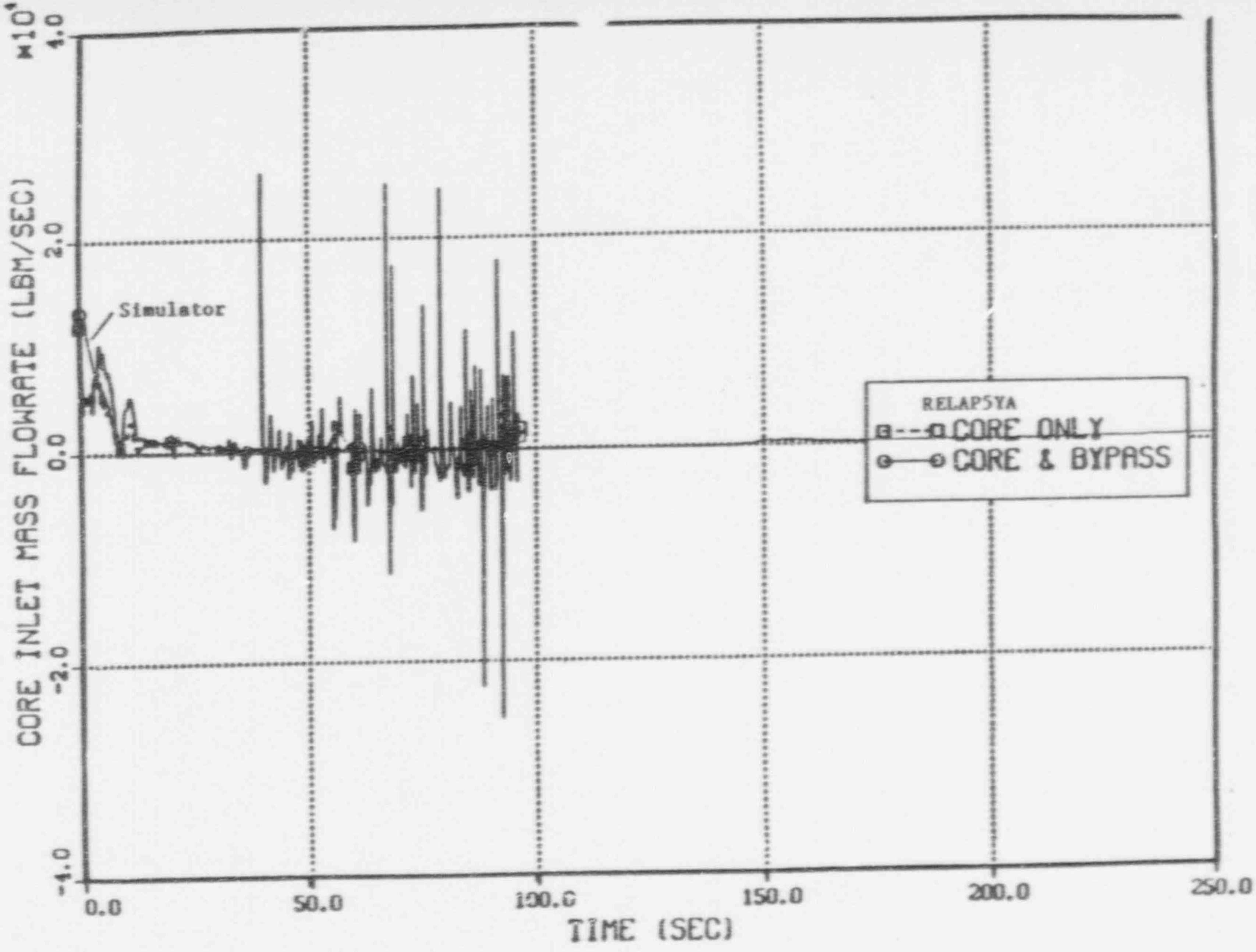


Figure 4-2.11 VY LB-LOCA Core Inlet and Bypass Inlet Flows

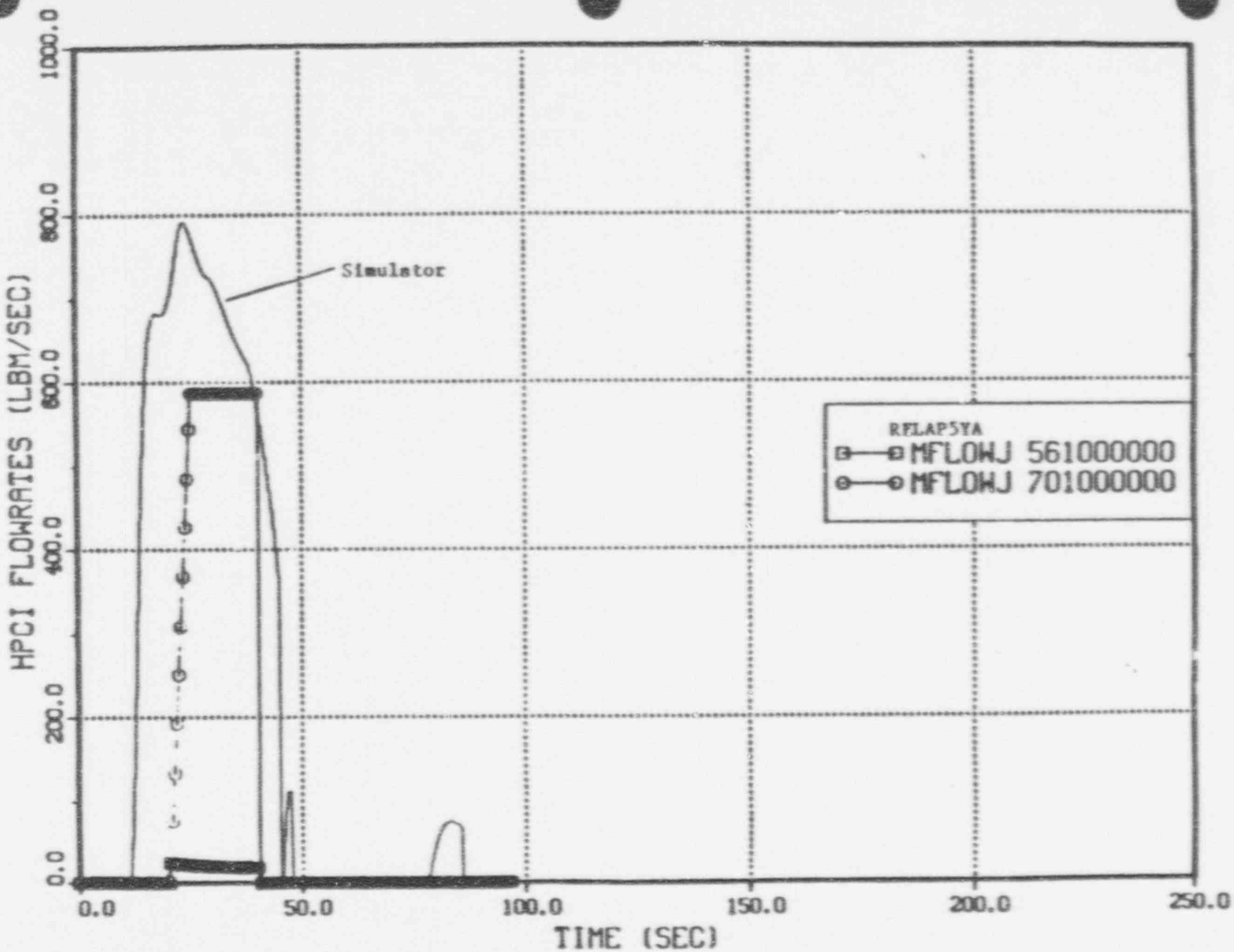


Figure 4-2.12 VY LB-LOCA RELAP5YA HPCI Flow

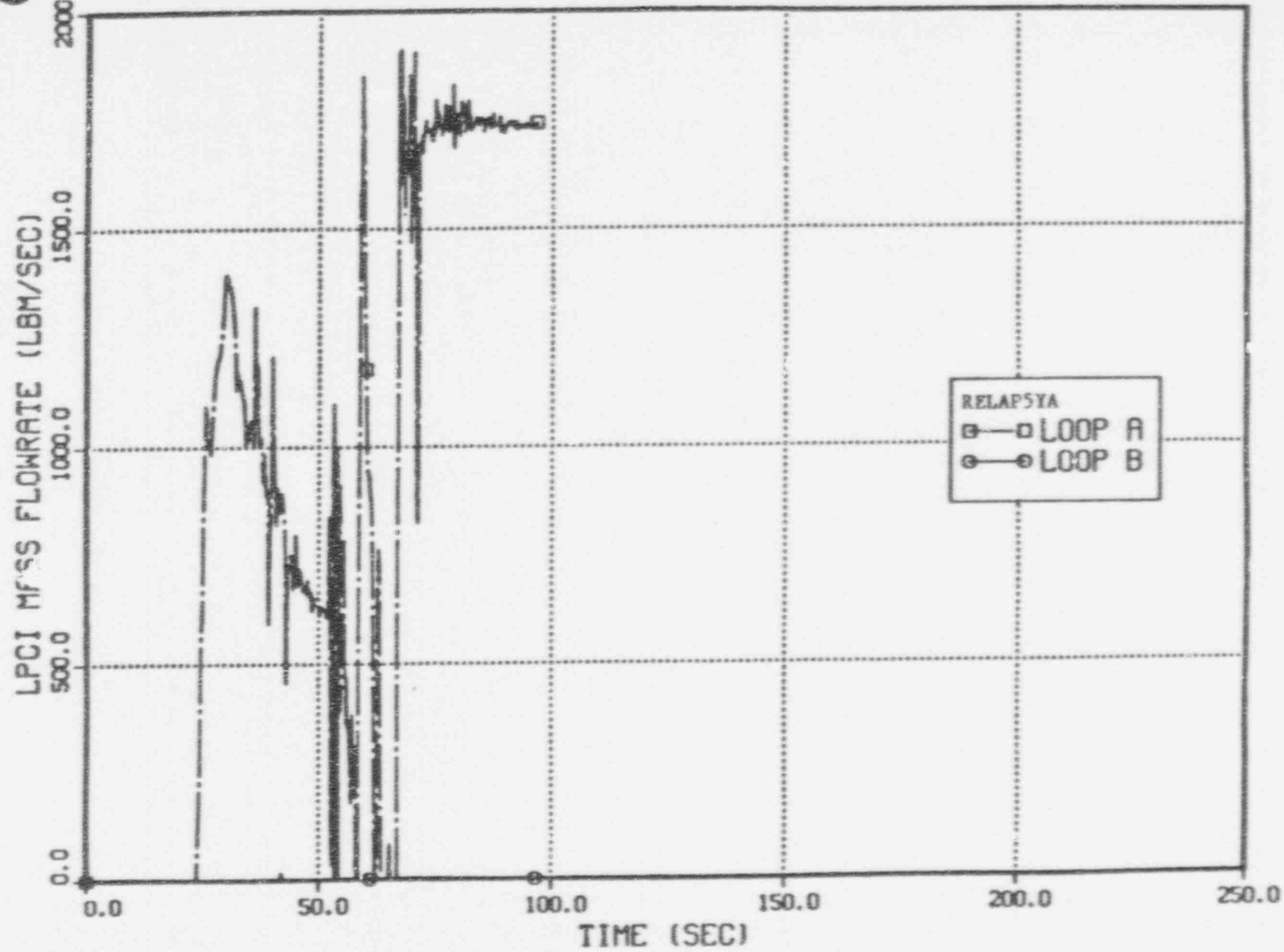


Figure 4-2.13 VT LB-LOCA RELAP5YA LPCI Pump Flow

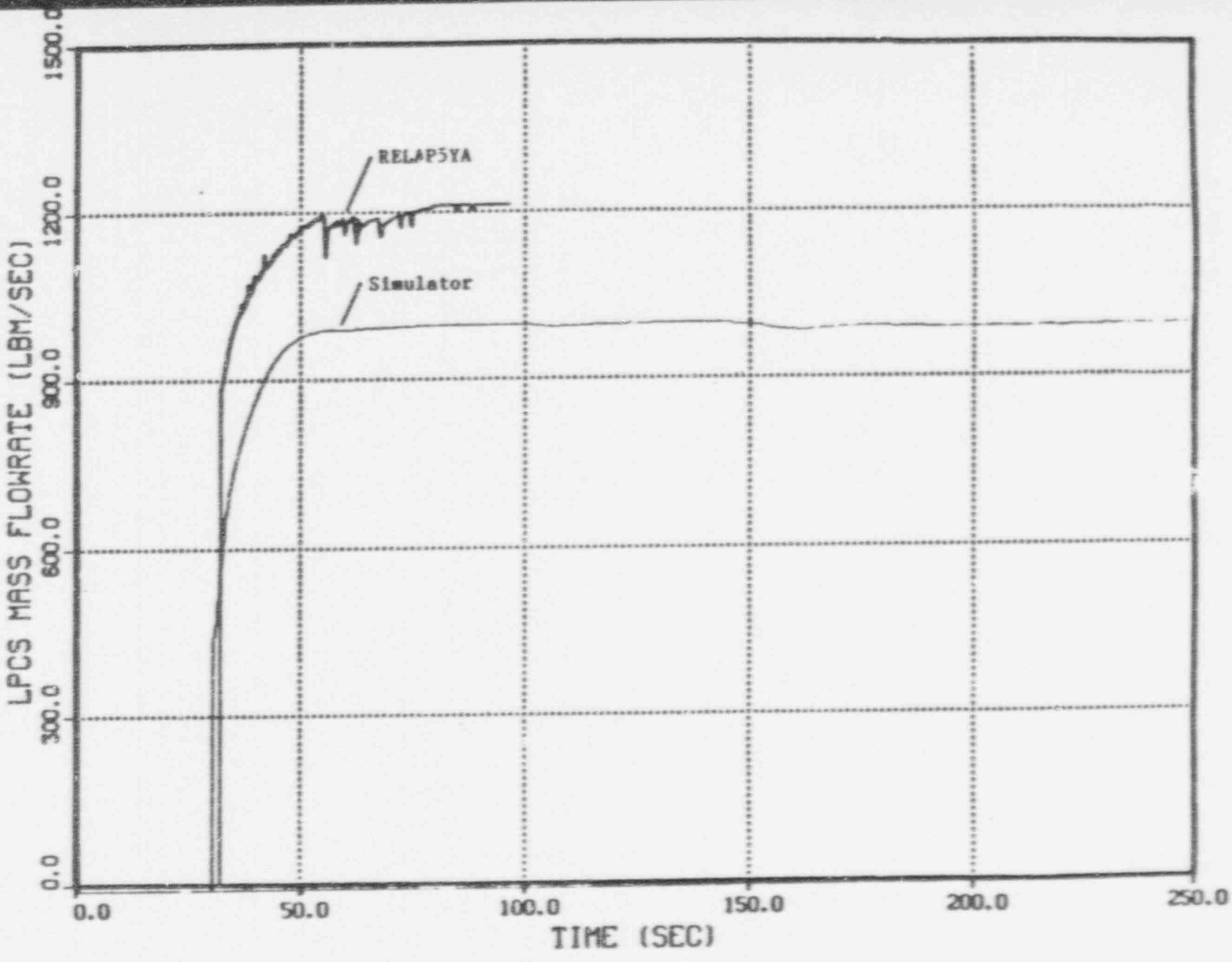


Figure 4-2.14 VY LB-LOCA LPCS Flow

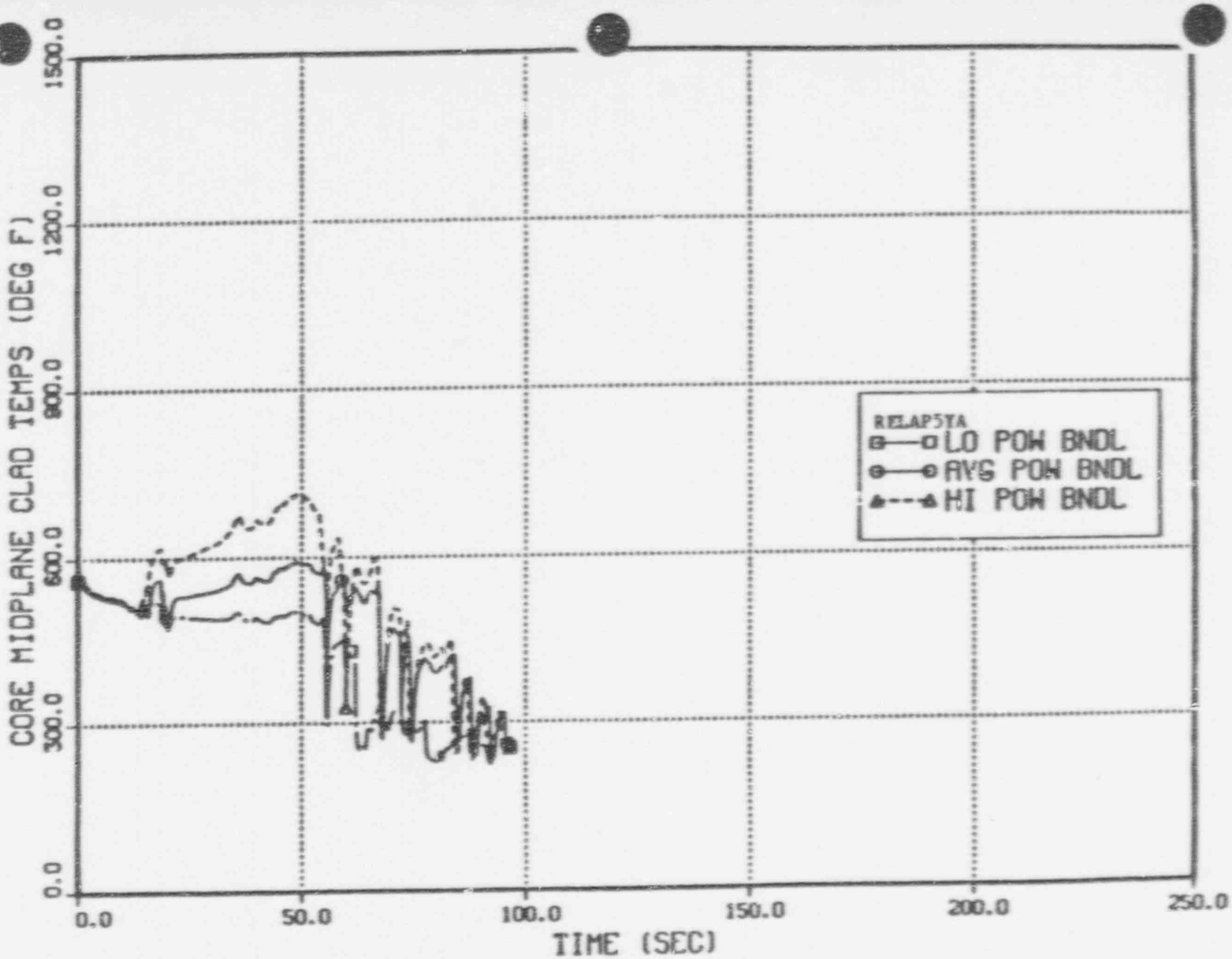


Figure 4-2.15 VY LB-LOCA Mid-Plane Elevation Clad Temperatures

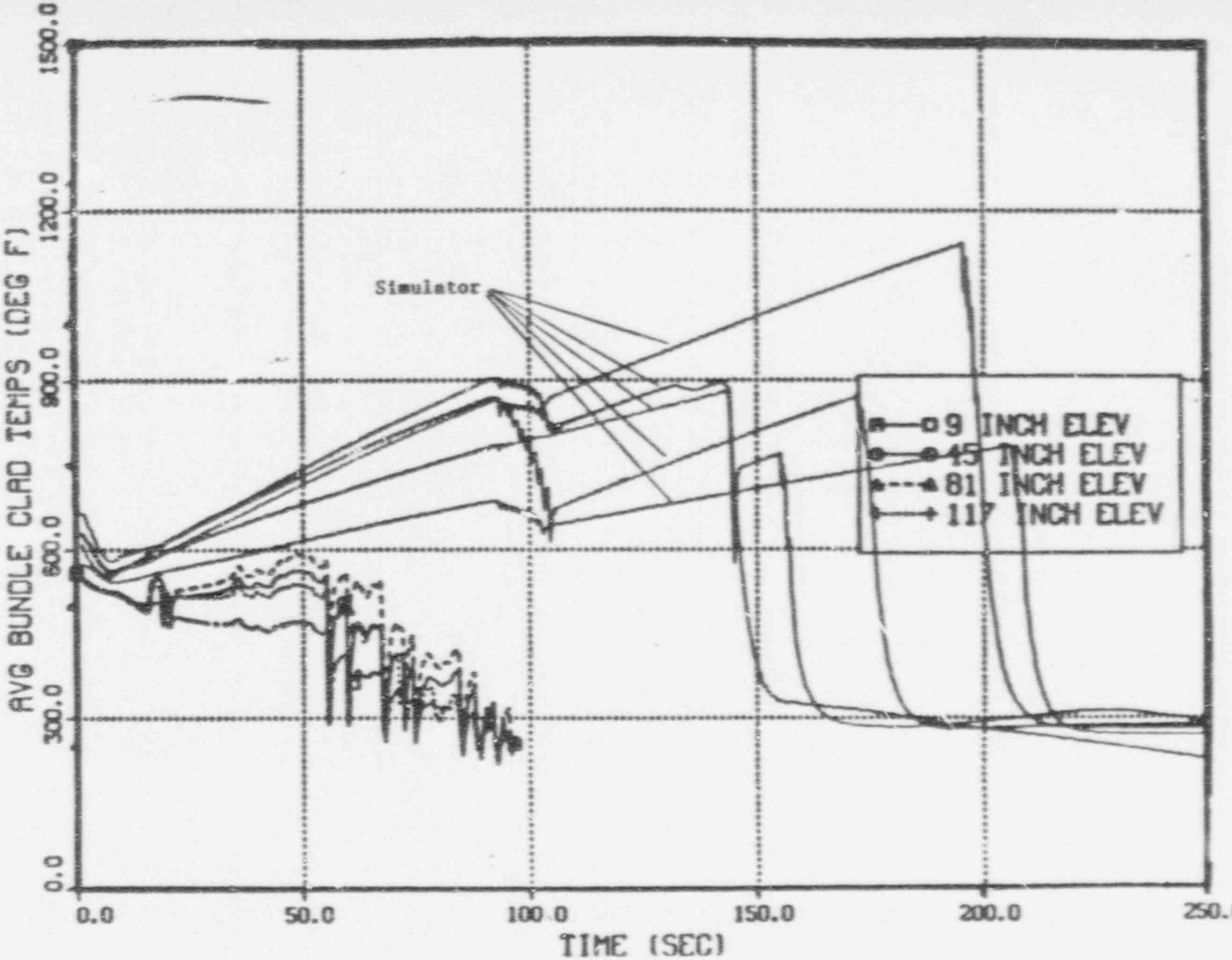


Figure 4-2.16 VT LB-LOCA Average Bundle Clad Temperatures

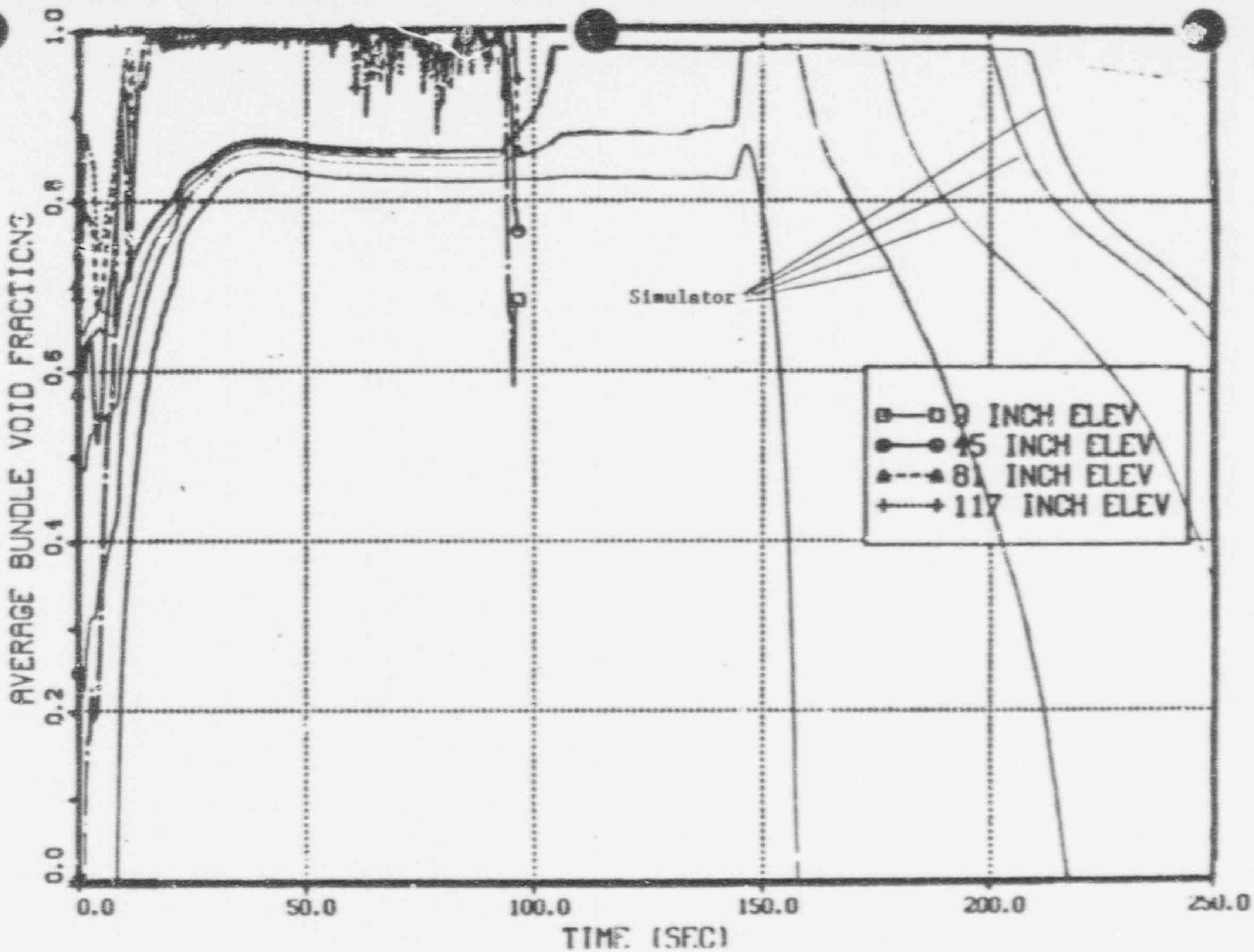


Figure 4-2.17 VT LB-LOCA Bundle Void Fractions

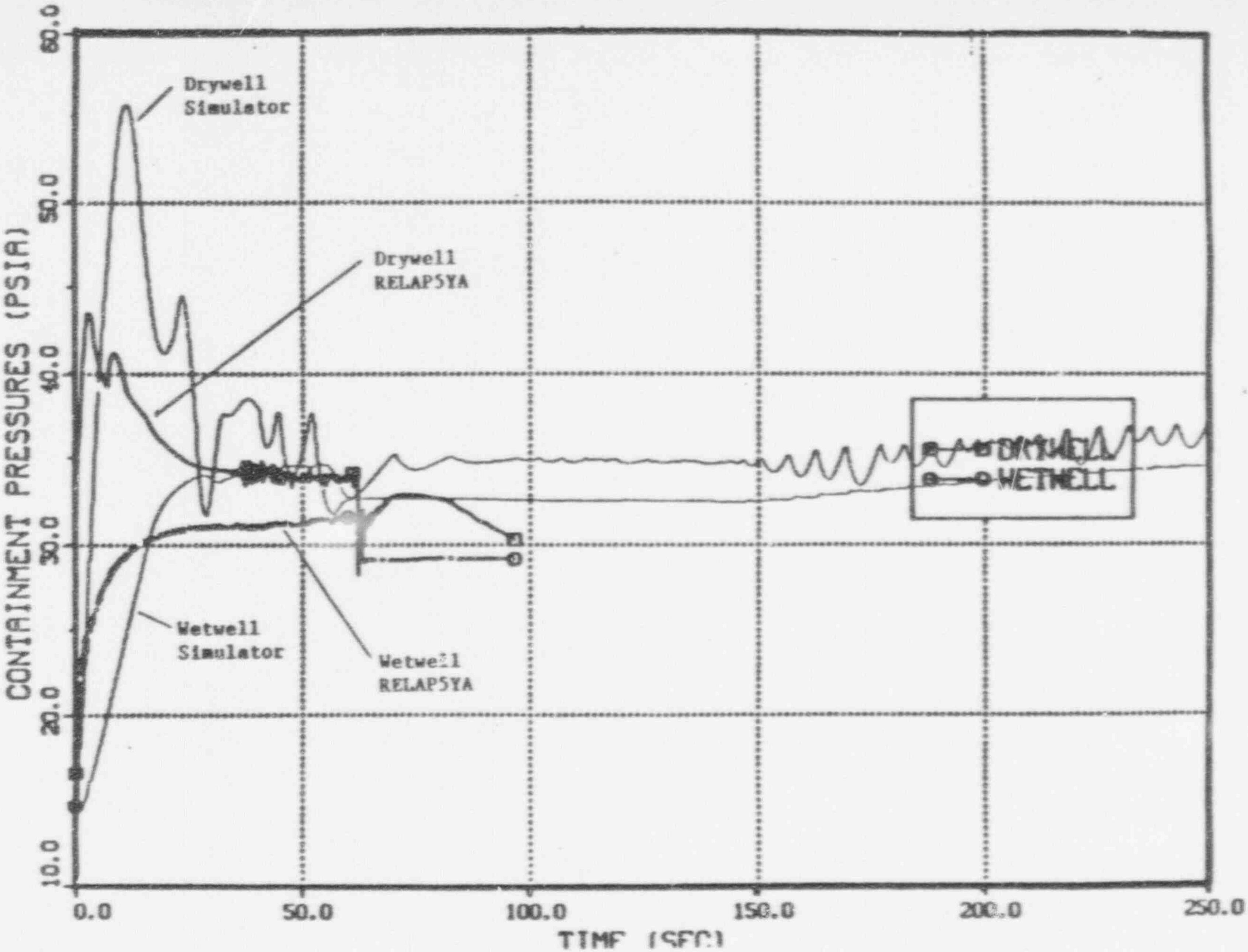


Figure 4-2.18 VT LB-LOCA Containment Pressures

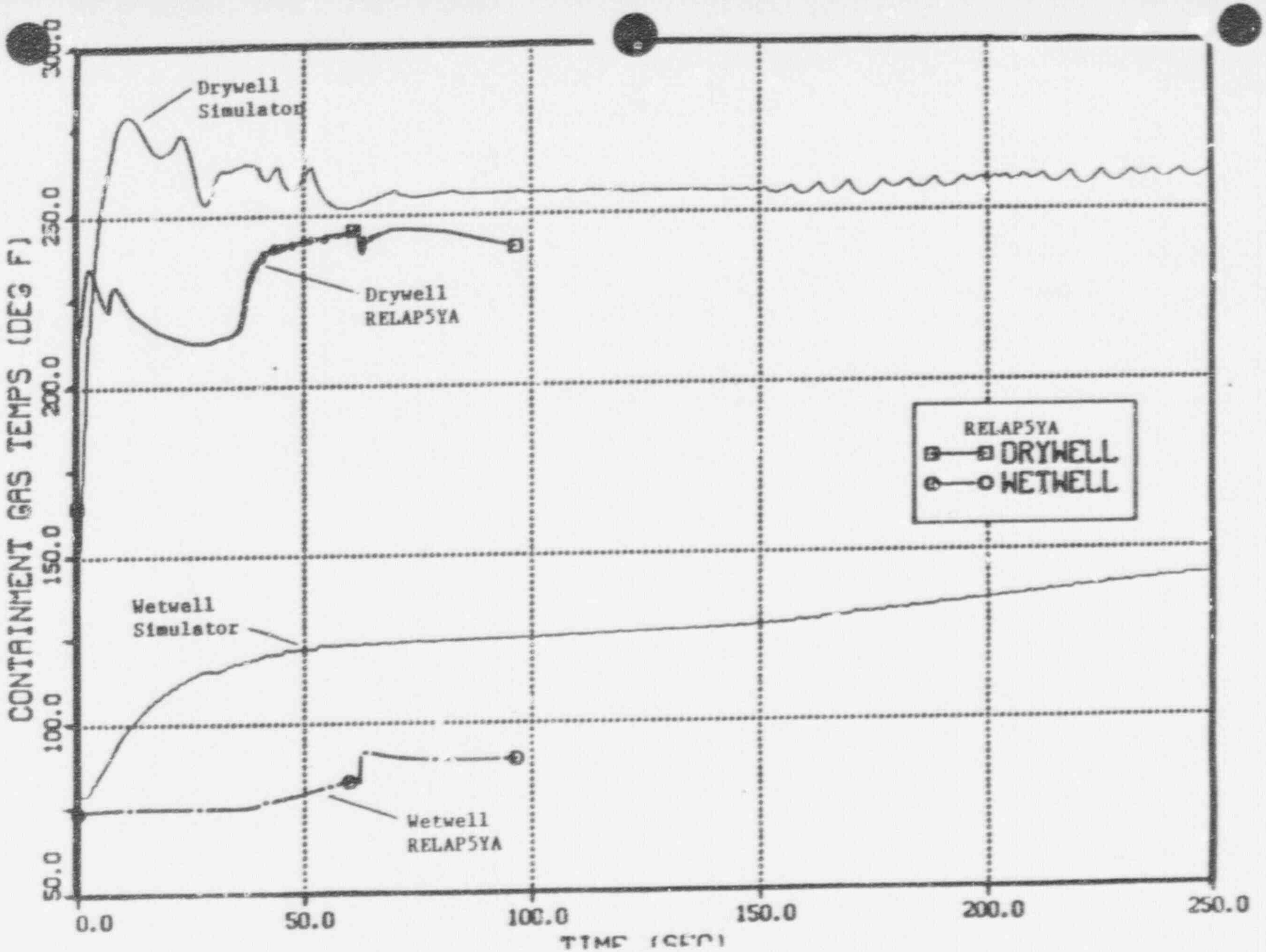


Figure 4-2.19 VT LB-LOCA Containment Gas Temperature

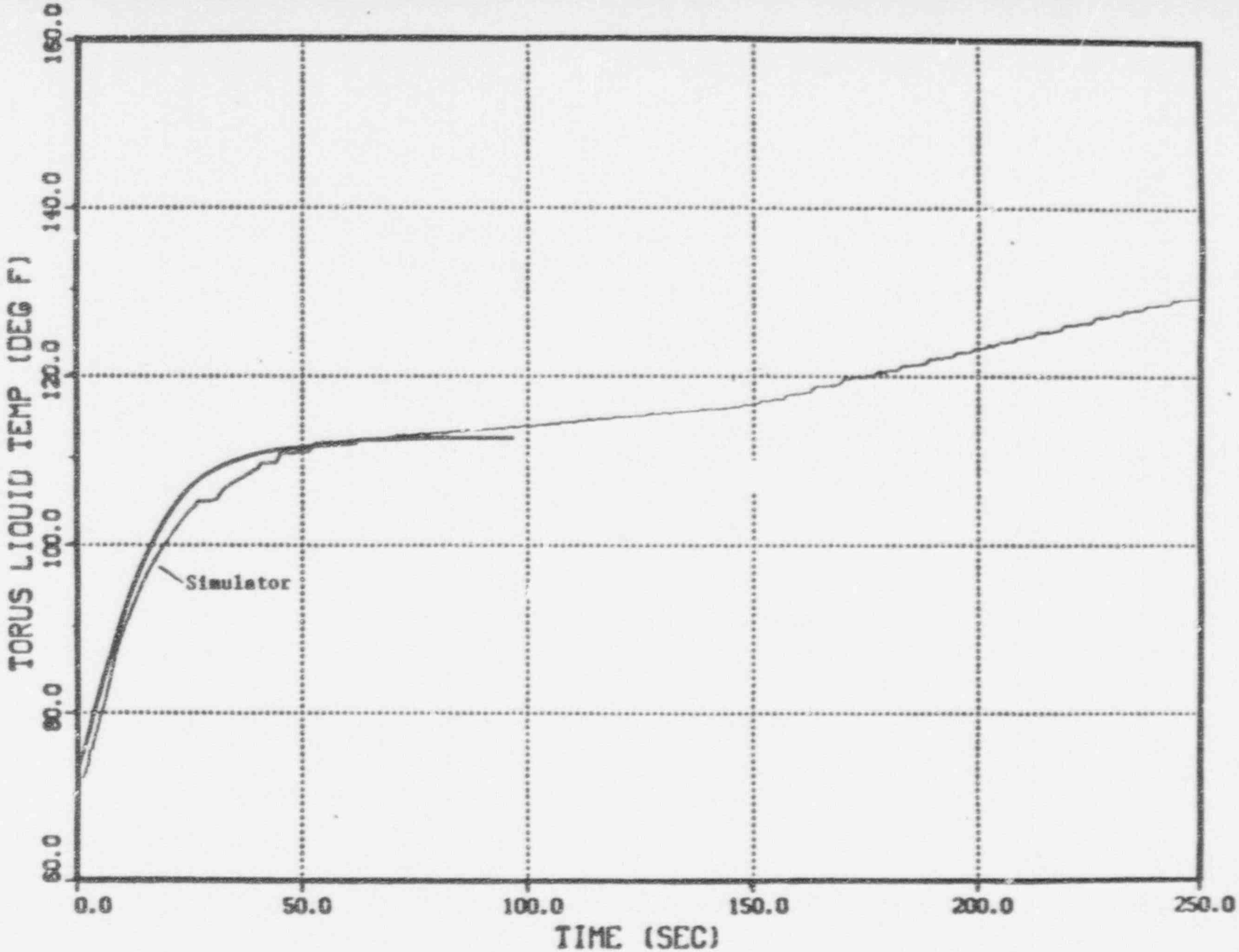


Figure 4-2.20 VY LB-LOCA Torus Liquid Temperature

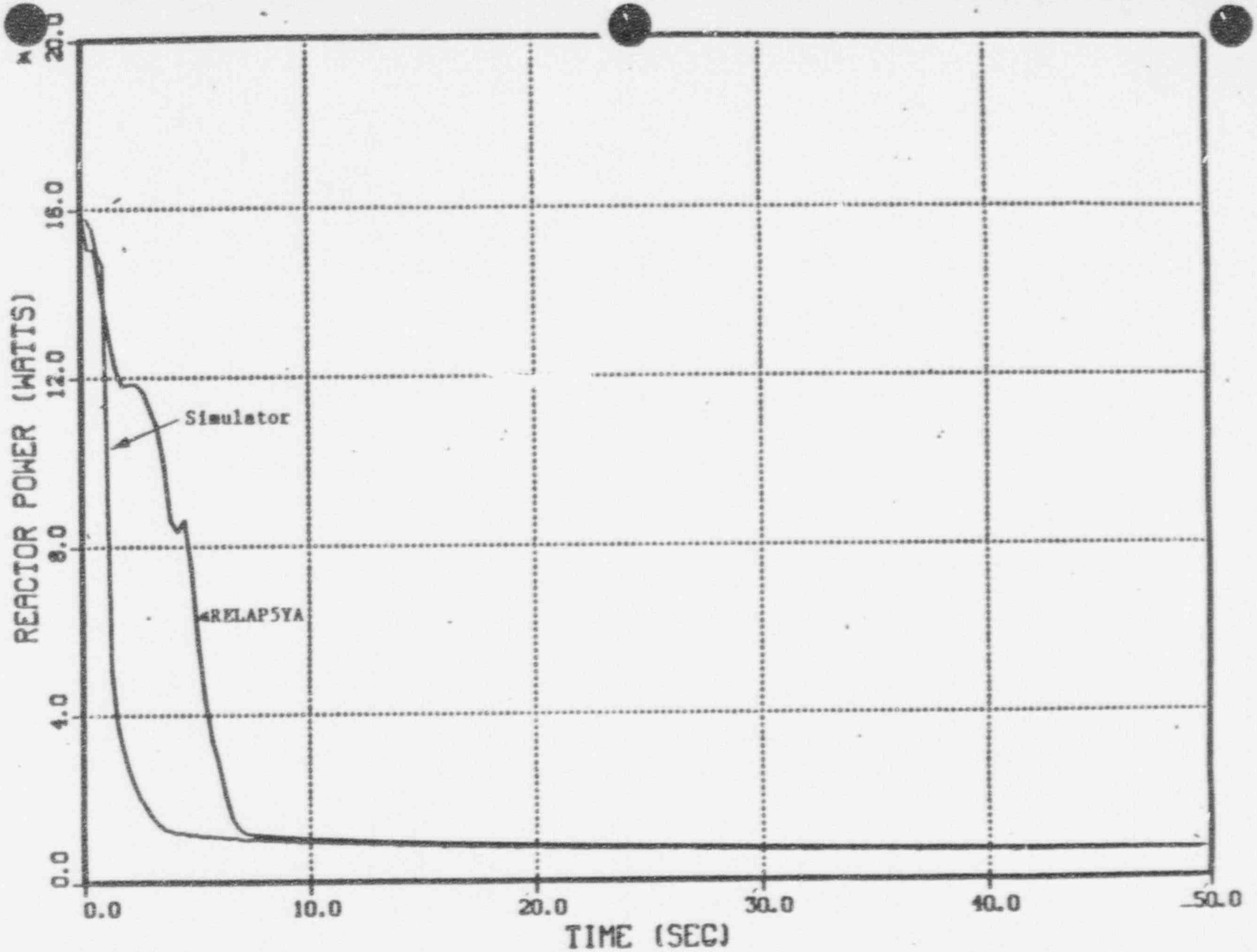


Figure 4-3.1 VY SB-LOCA Core Thermal Power

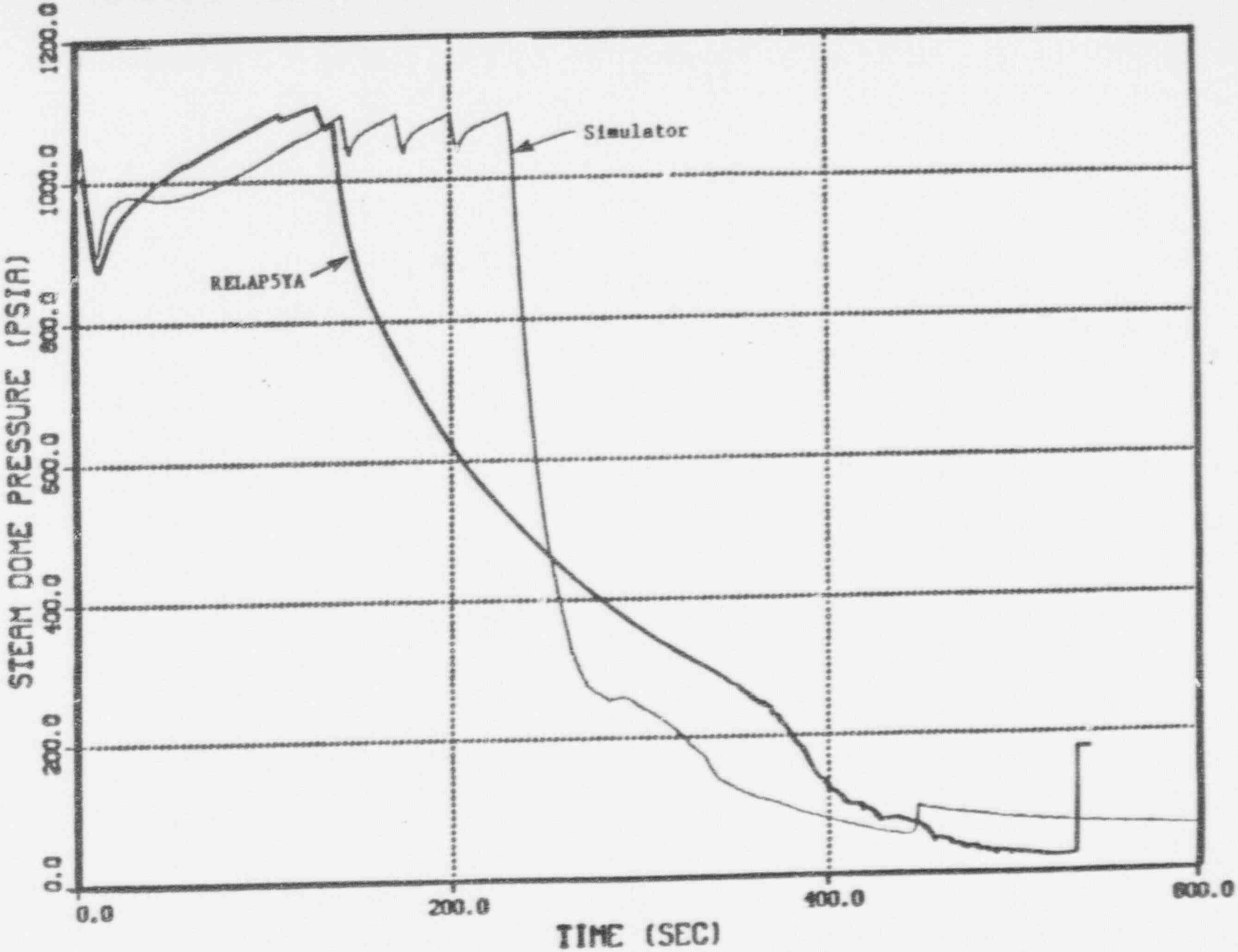


Figure 4-3.2 VY SB-LOCA Reactor Vessel Steam Dome Pressure

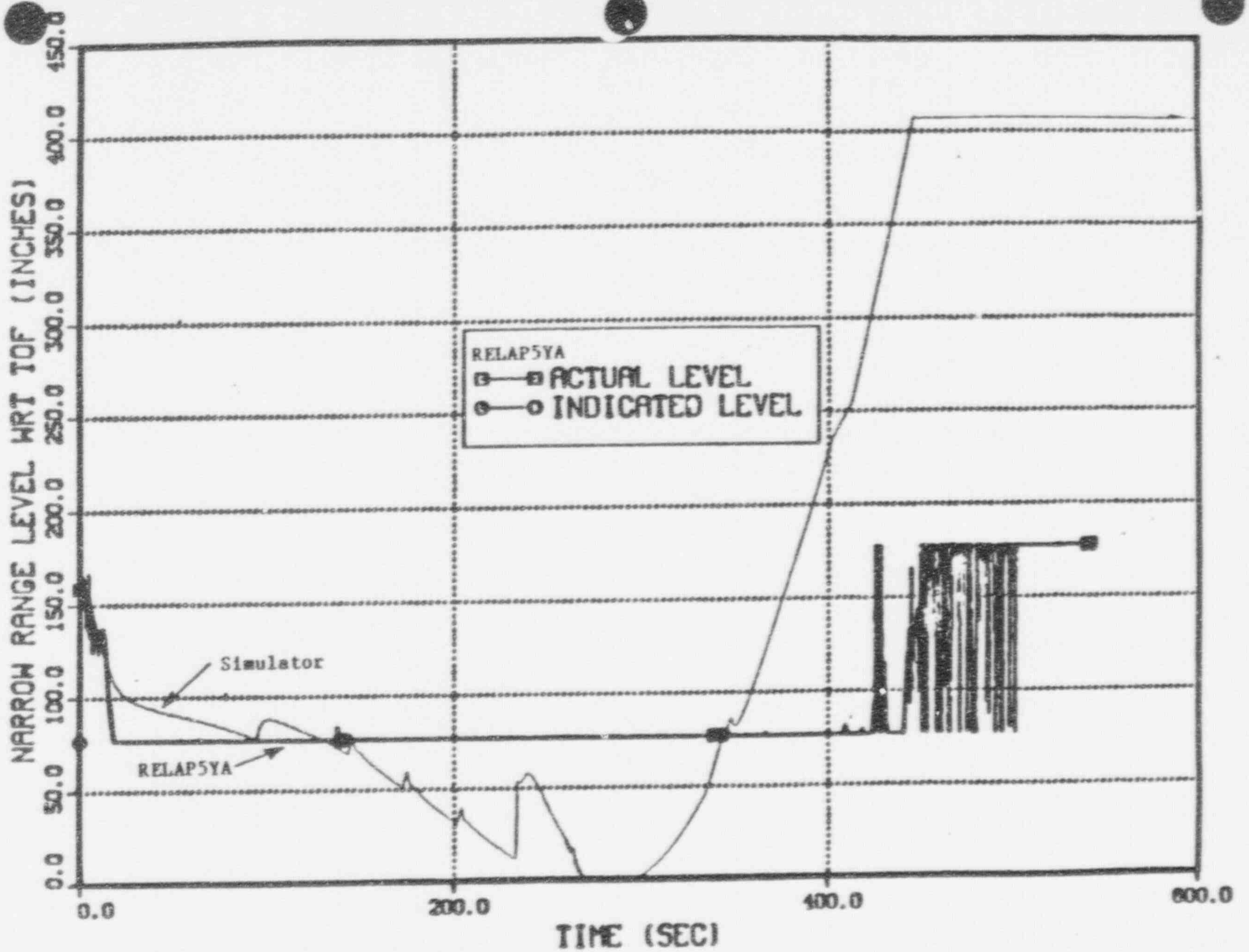


Figure 4-3.3 VY SB-LOCA Reactor Vessel Narrow Range Water Level

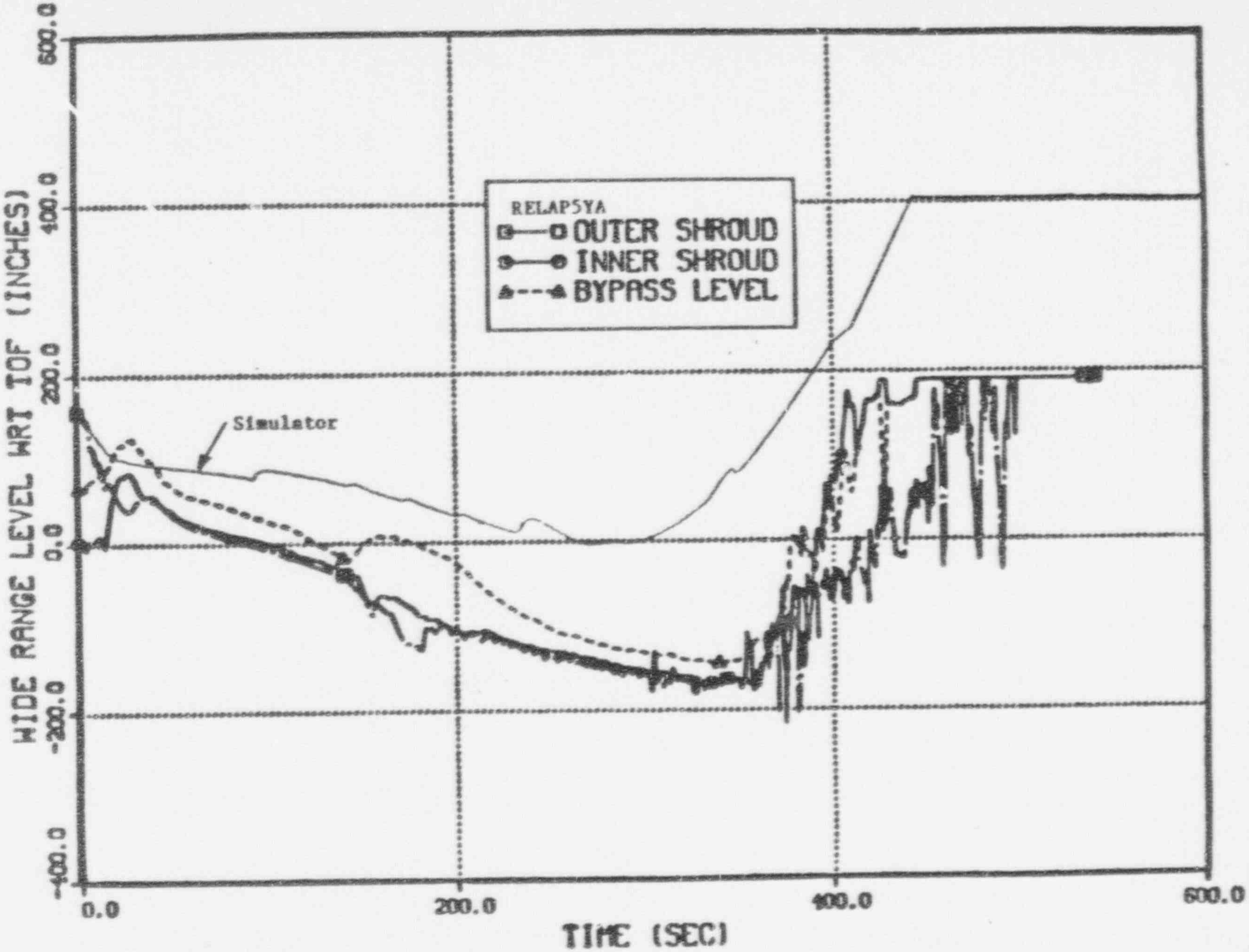


Figure 4-3.4 VY SB-LOCA Reactor Vessel Wide Range Water Level

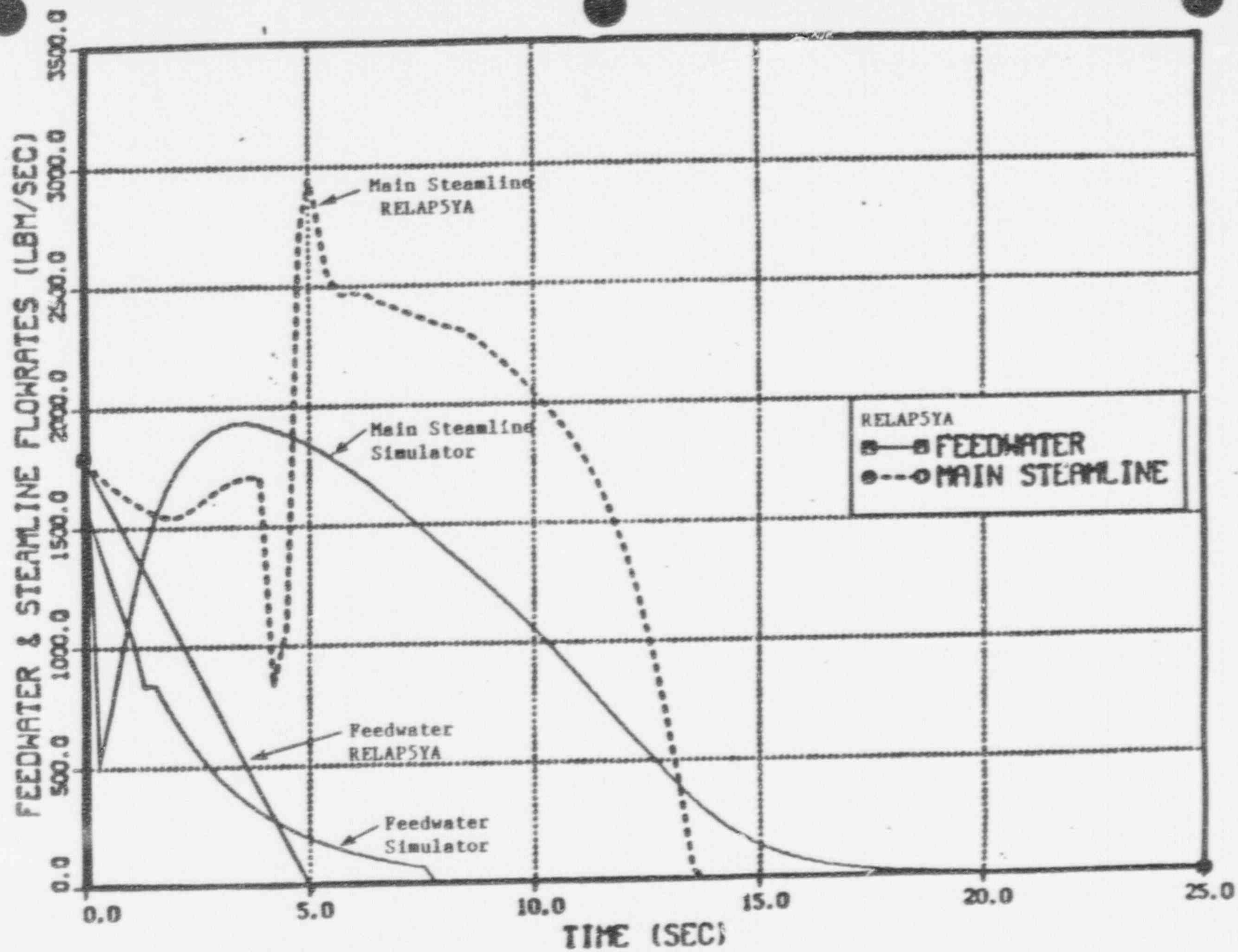


Figure 4-3.5 VY SB-LOCA Feedwater and Steam Line Flows

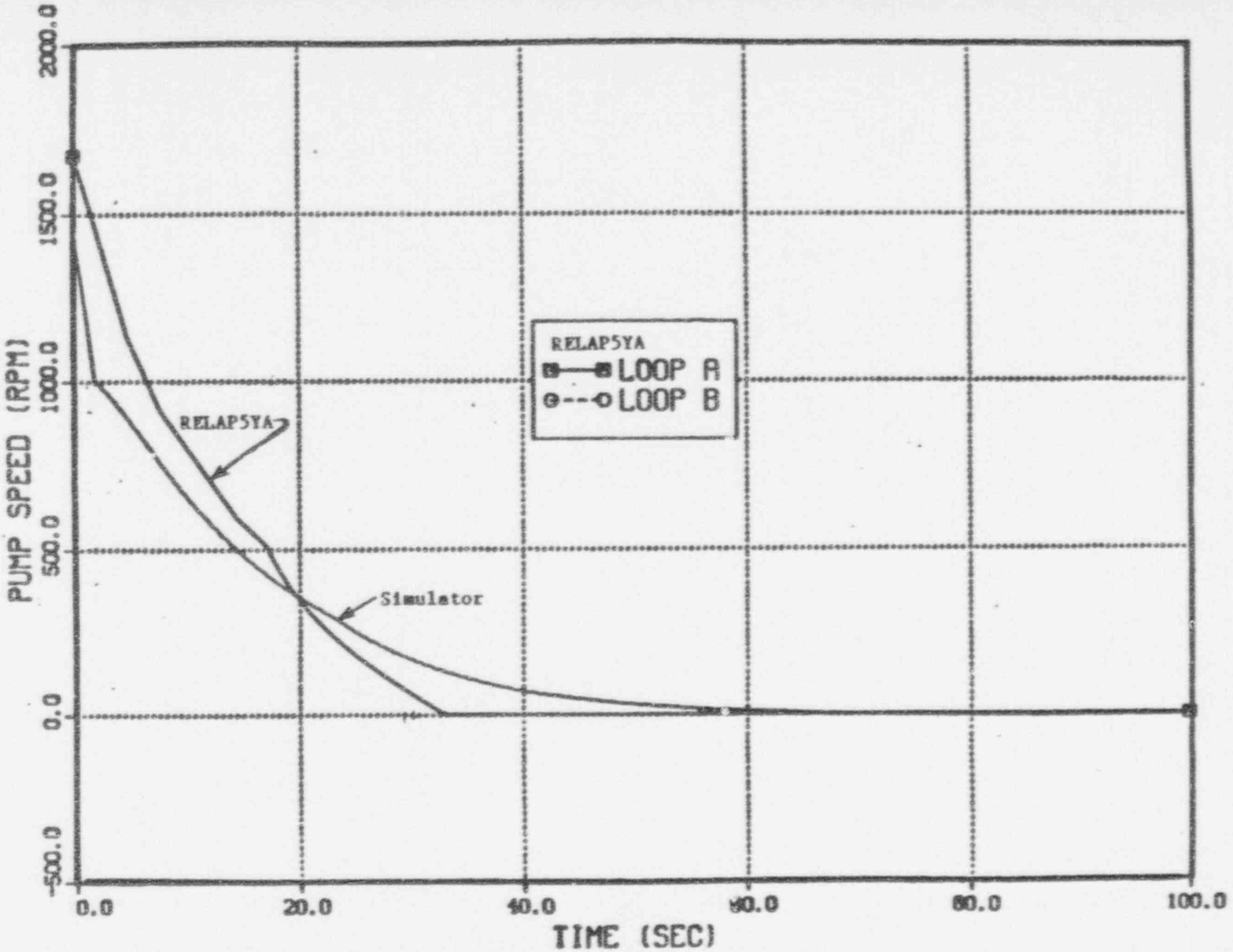


Figure 4-3.6 VY SB-LOCA Reactor Recirculation Loop Pump Speed

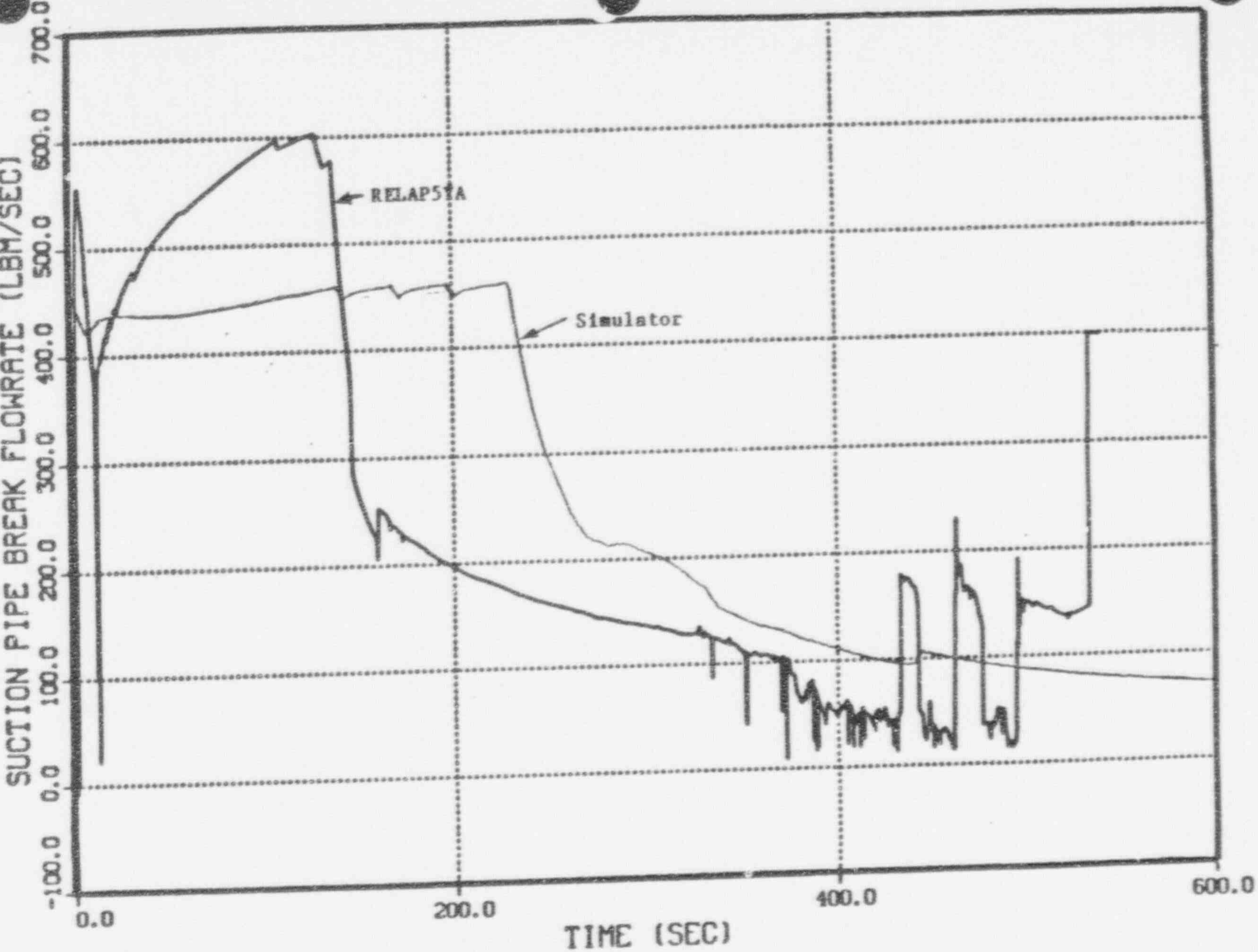


Figure 4-3.7 VY SB-LOCA Break Flow

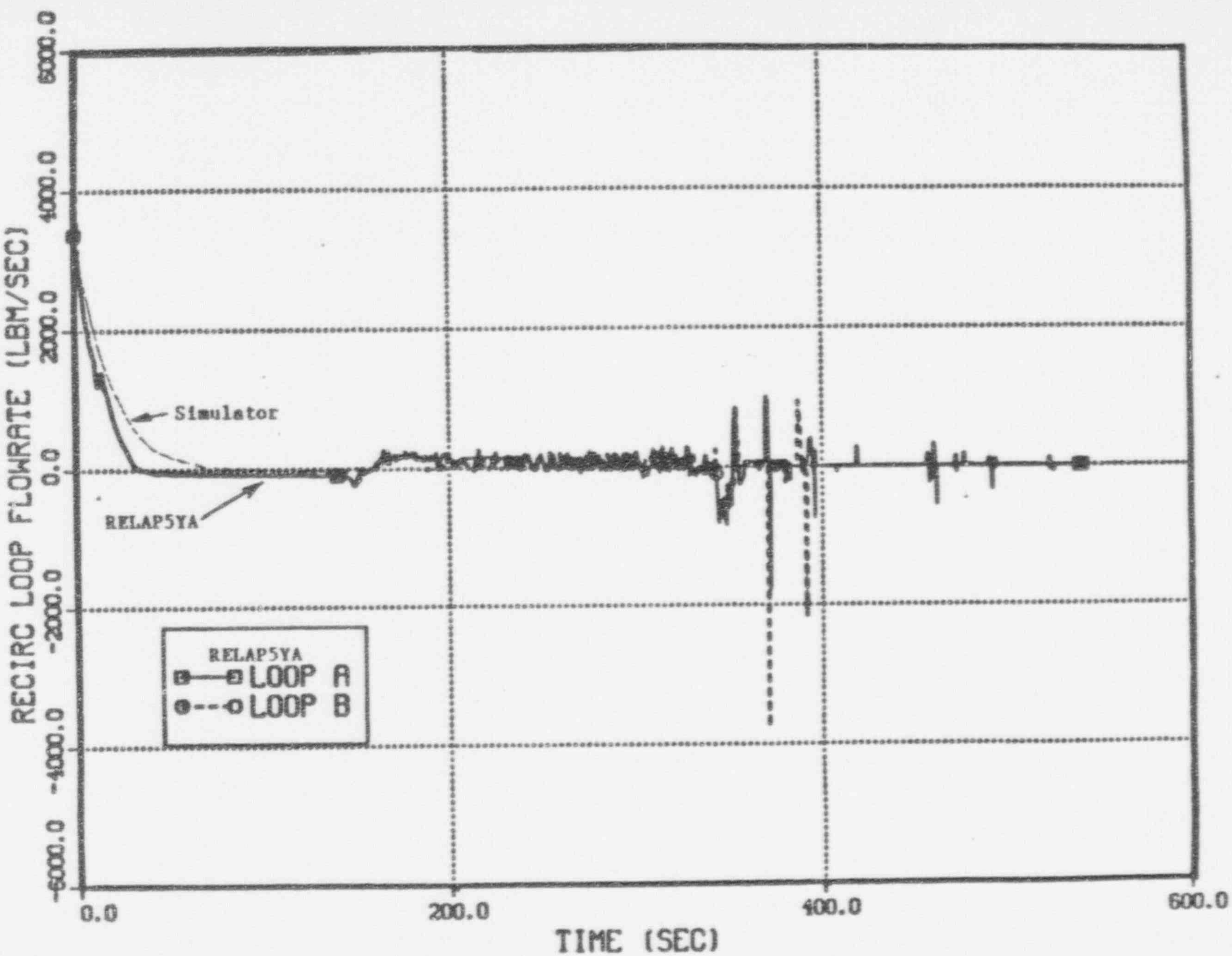


Figure 4-3.8 VY SB-LOCA Reactor Recirculation Loop Flows

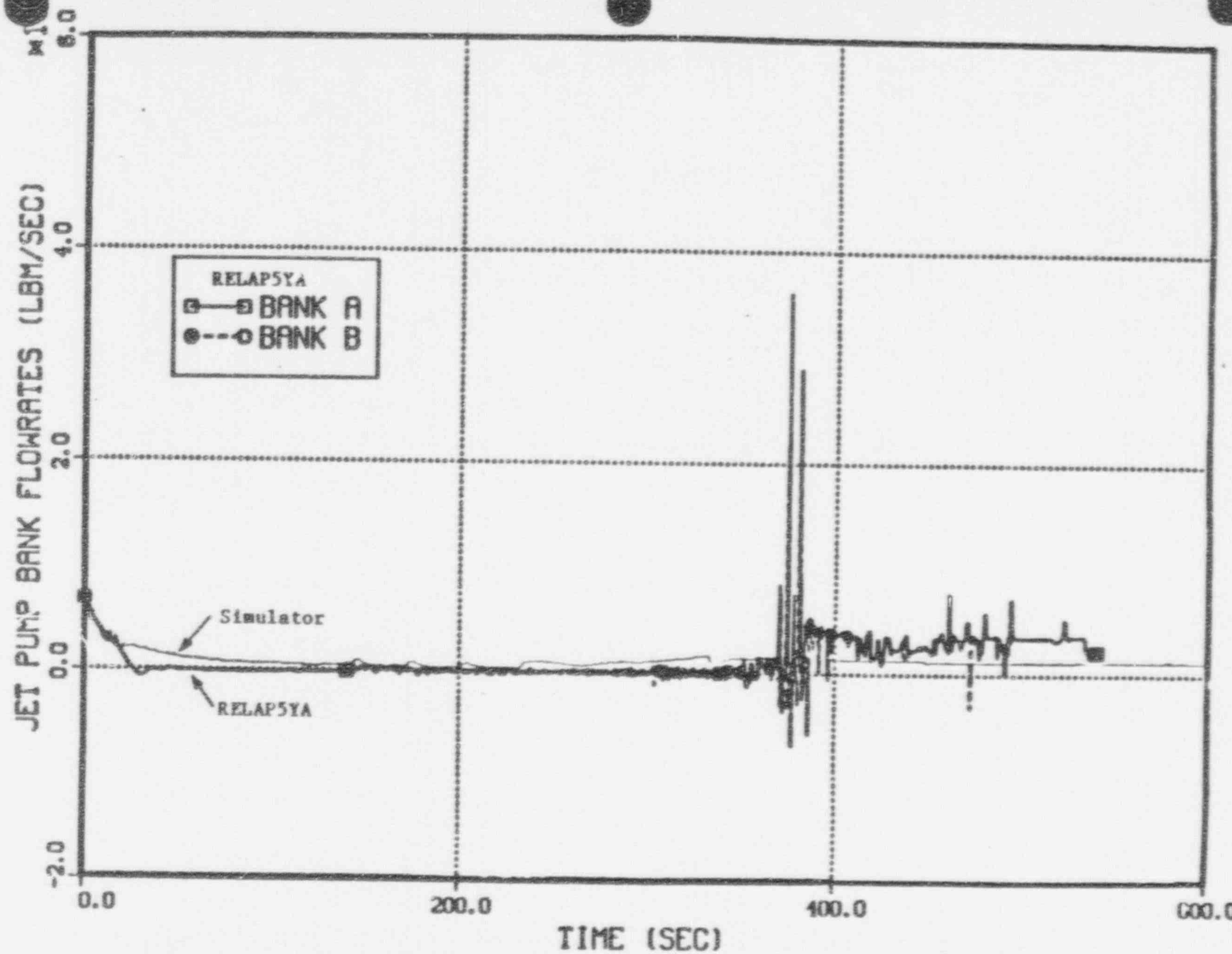


Figure 4-3.9 VY SB-LOCA Jet Pump Flows

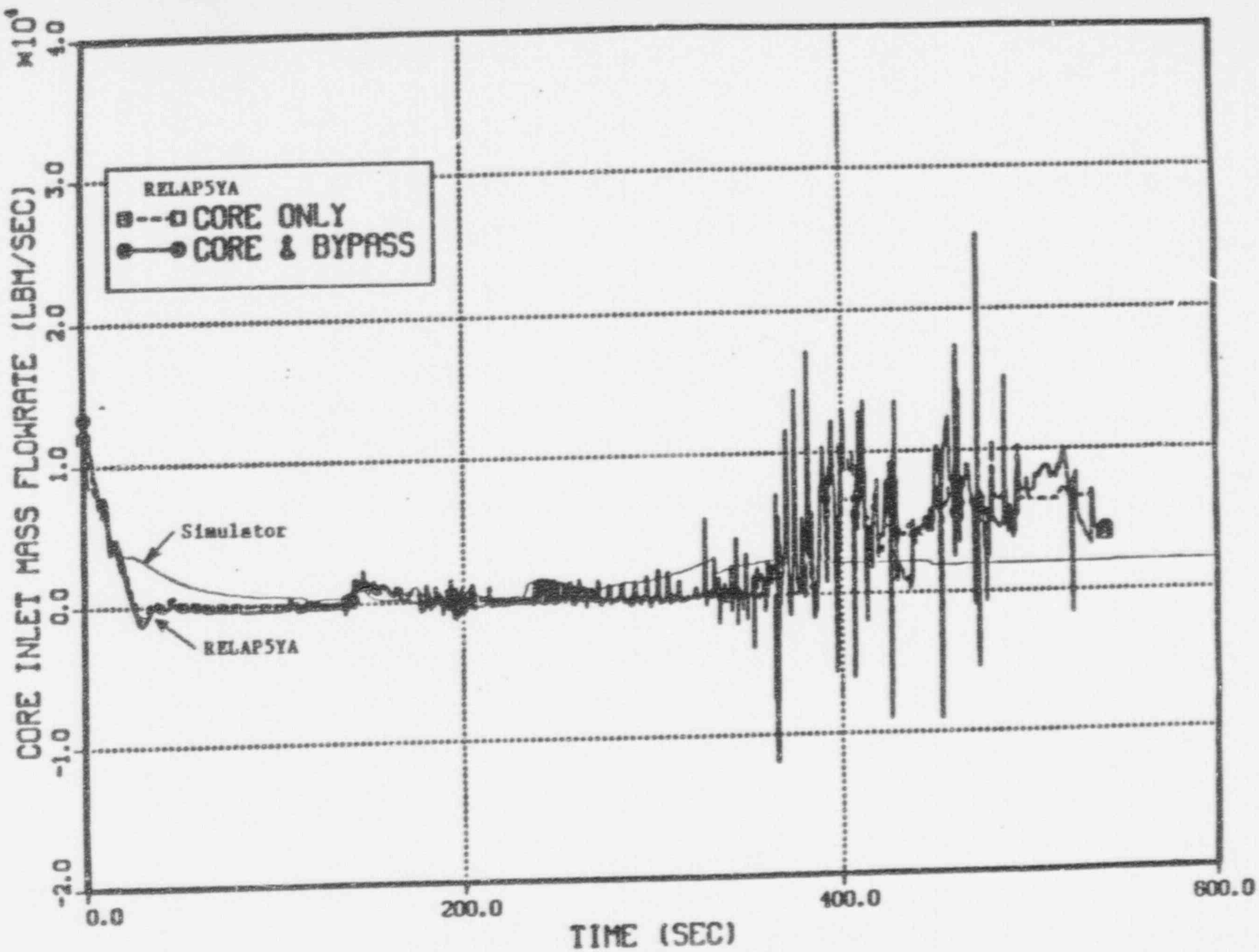


Figure 4-3.10 VY SB-LOCA Core Inlet and Bypass Inlet Flows

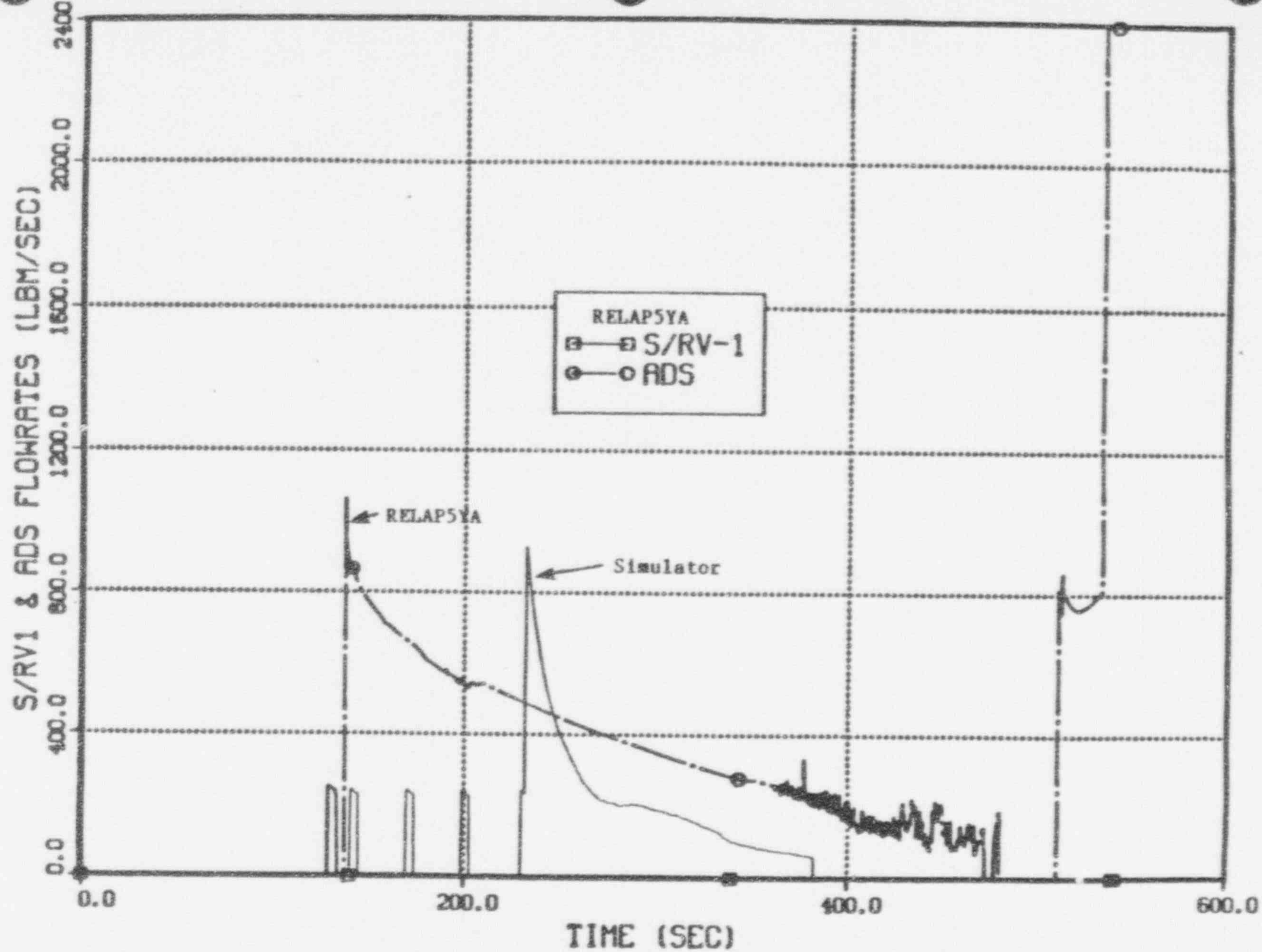


Figure 4-3.11 VY SB-LOCA Safety/Relief Valves and ADS Flows

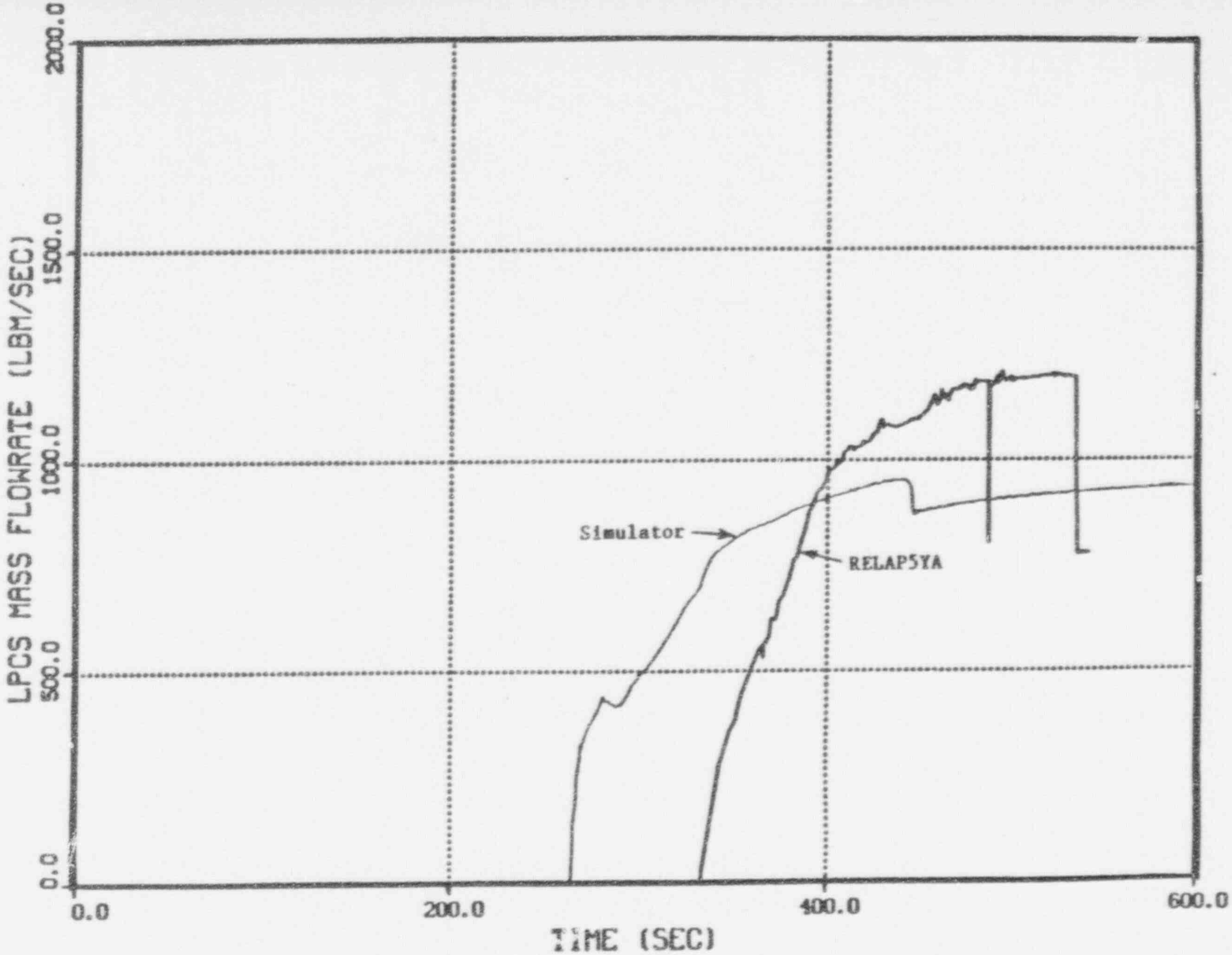


Figure 4-3.12 VY SB-LOCA LPCS Mass Flow

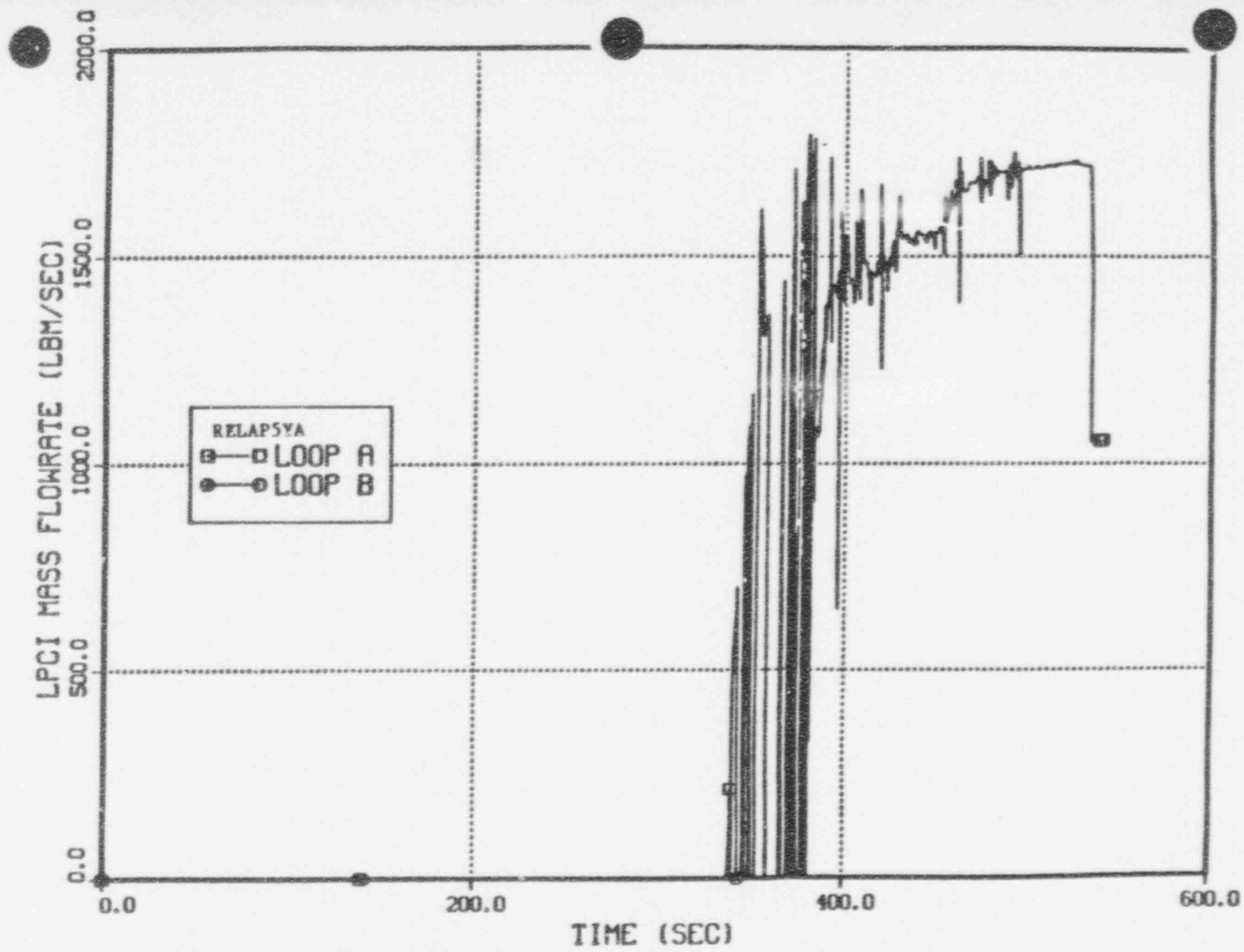


Figure 4-3.13 VY SB-LOCA LPCI Mass Flows

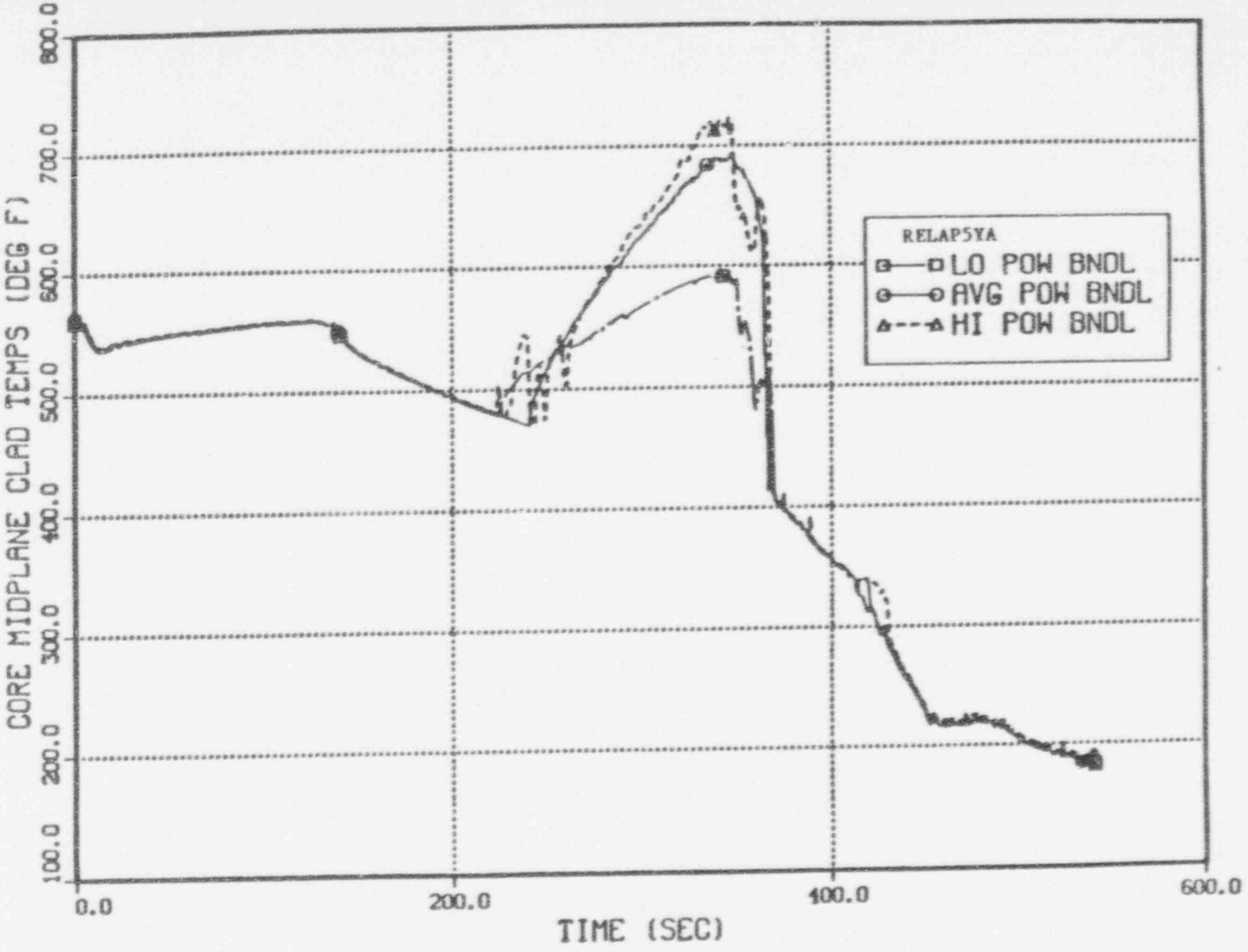


Figure 4-3.14 VY SB-LOCA Mid-Plane Elevation Clad Temperatures

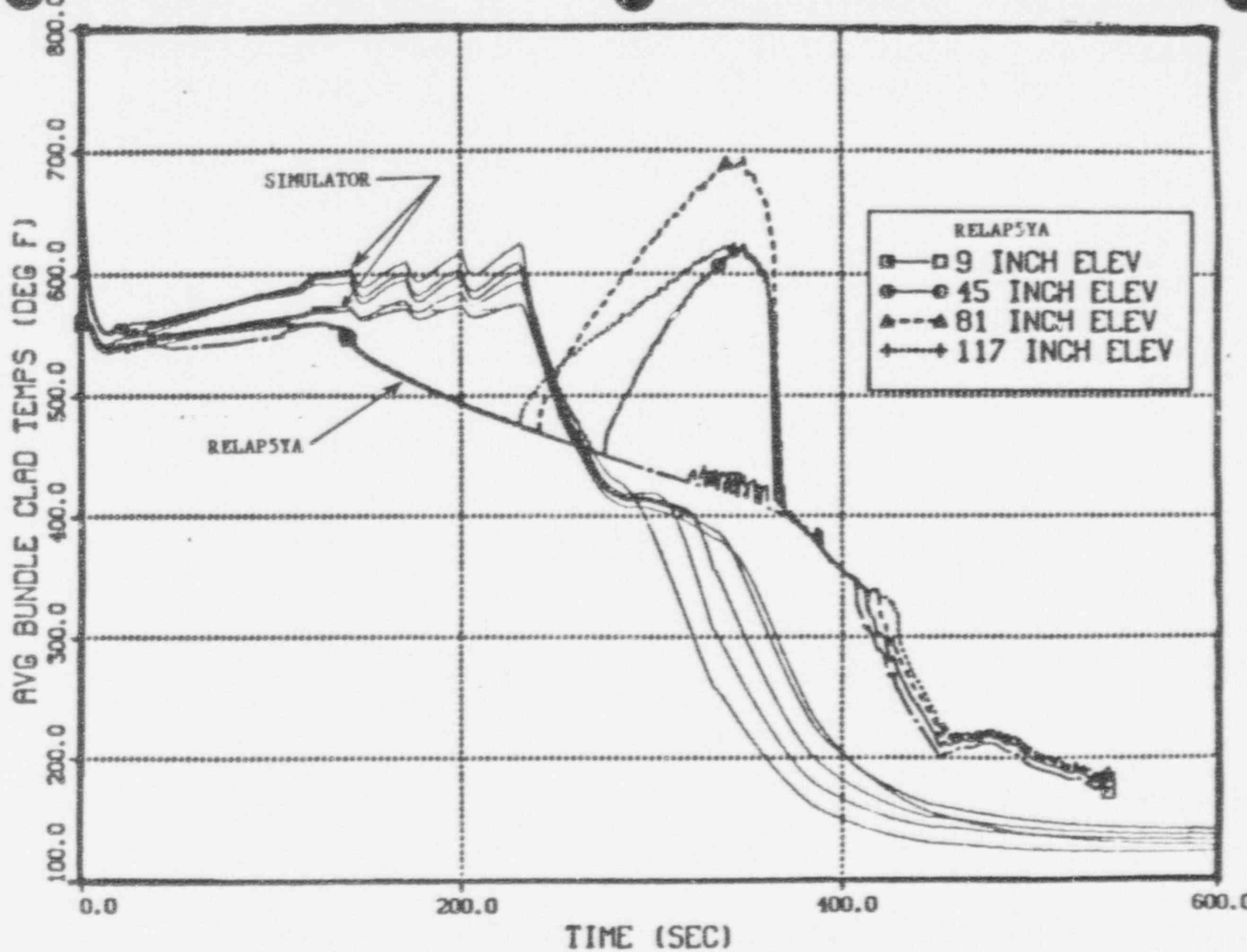


Figure 4-3.15 VT SB-LOCA Average Bundle Clad Temperatures

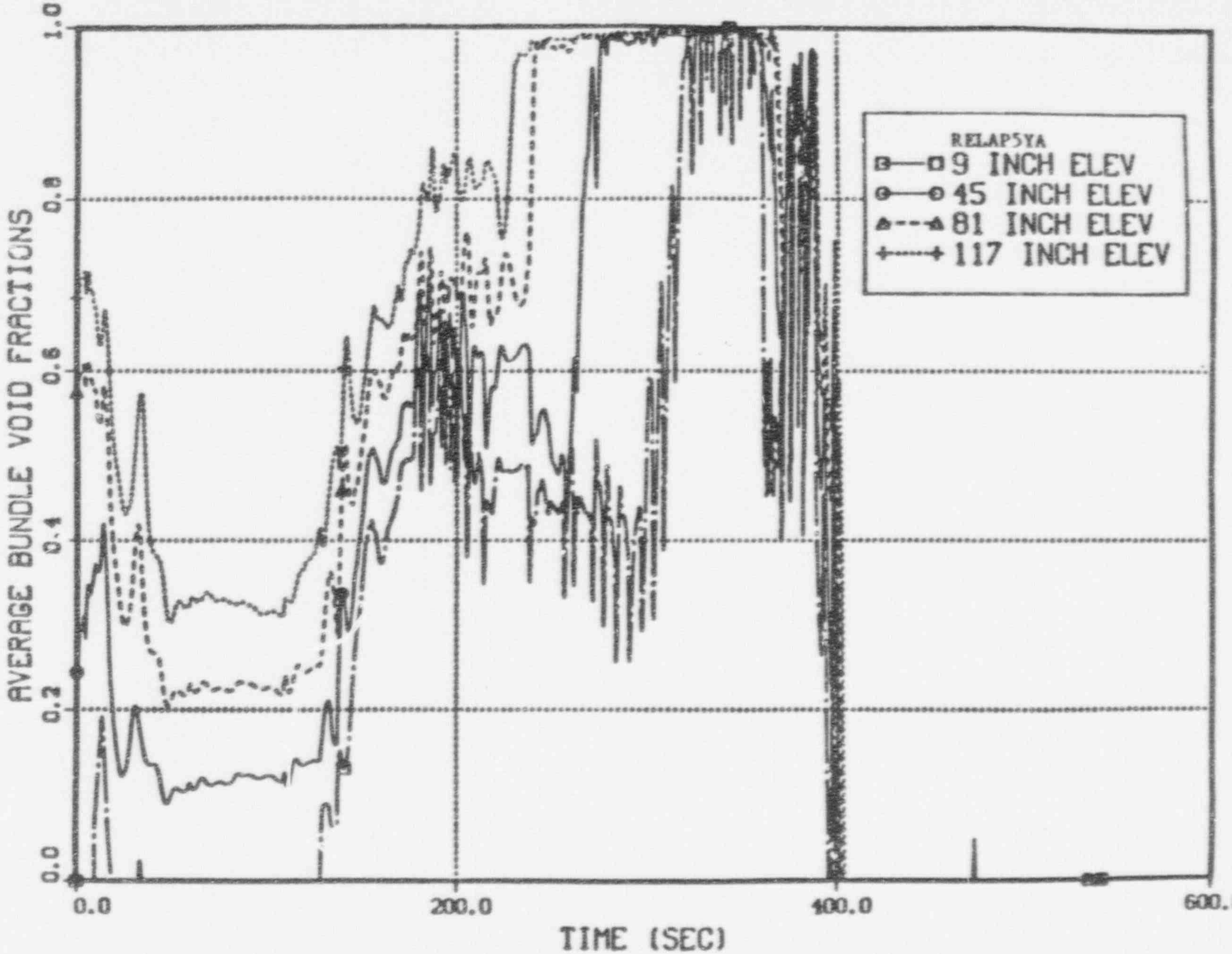


Figure 4-3.16 VY SB-LOCA RELAP5YA Average Bundle Void Fractions

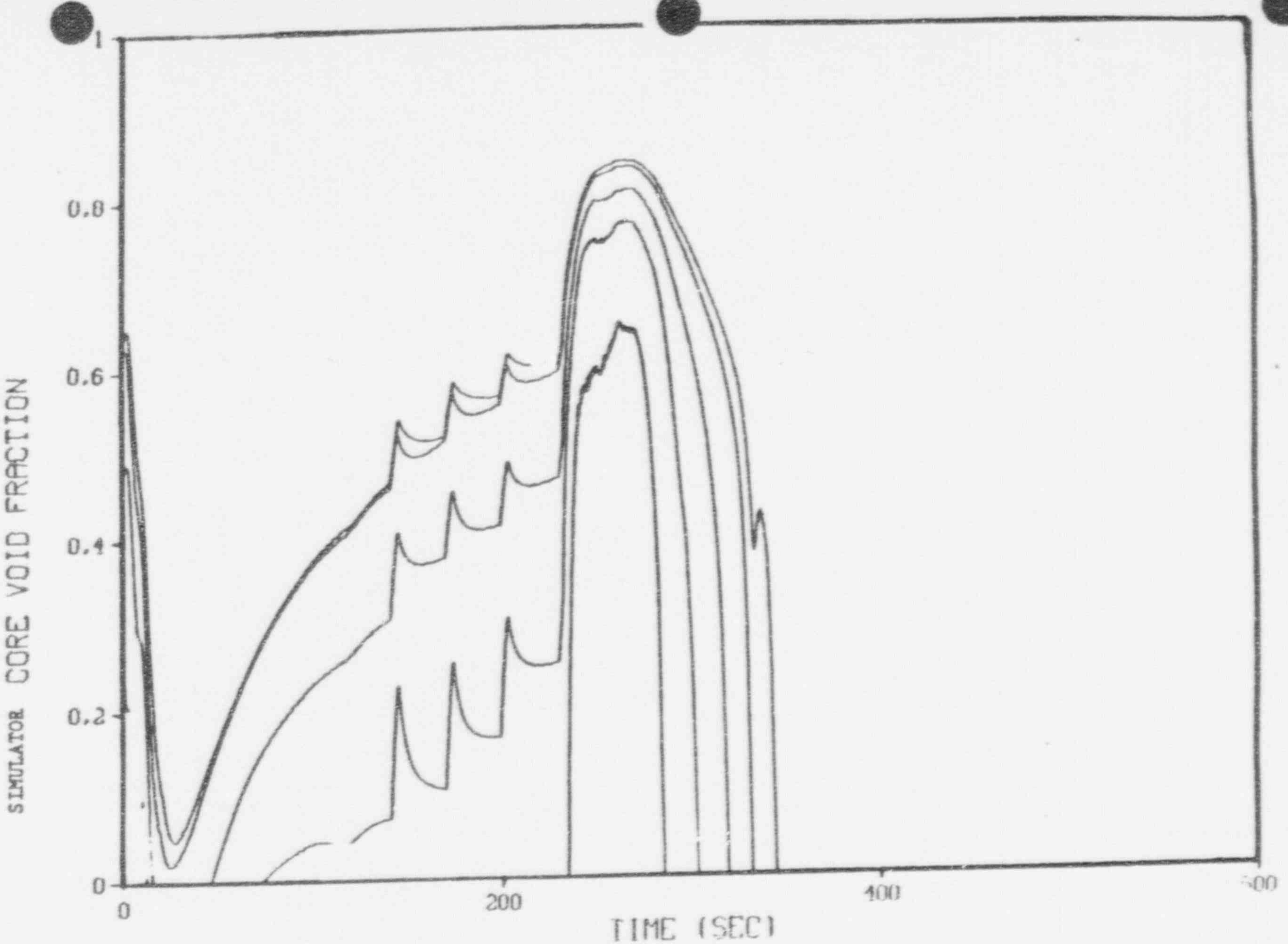


Figure 4-3.17 VY SB-LOCA Simulator Average Bundle Void Fractions

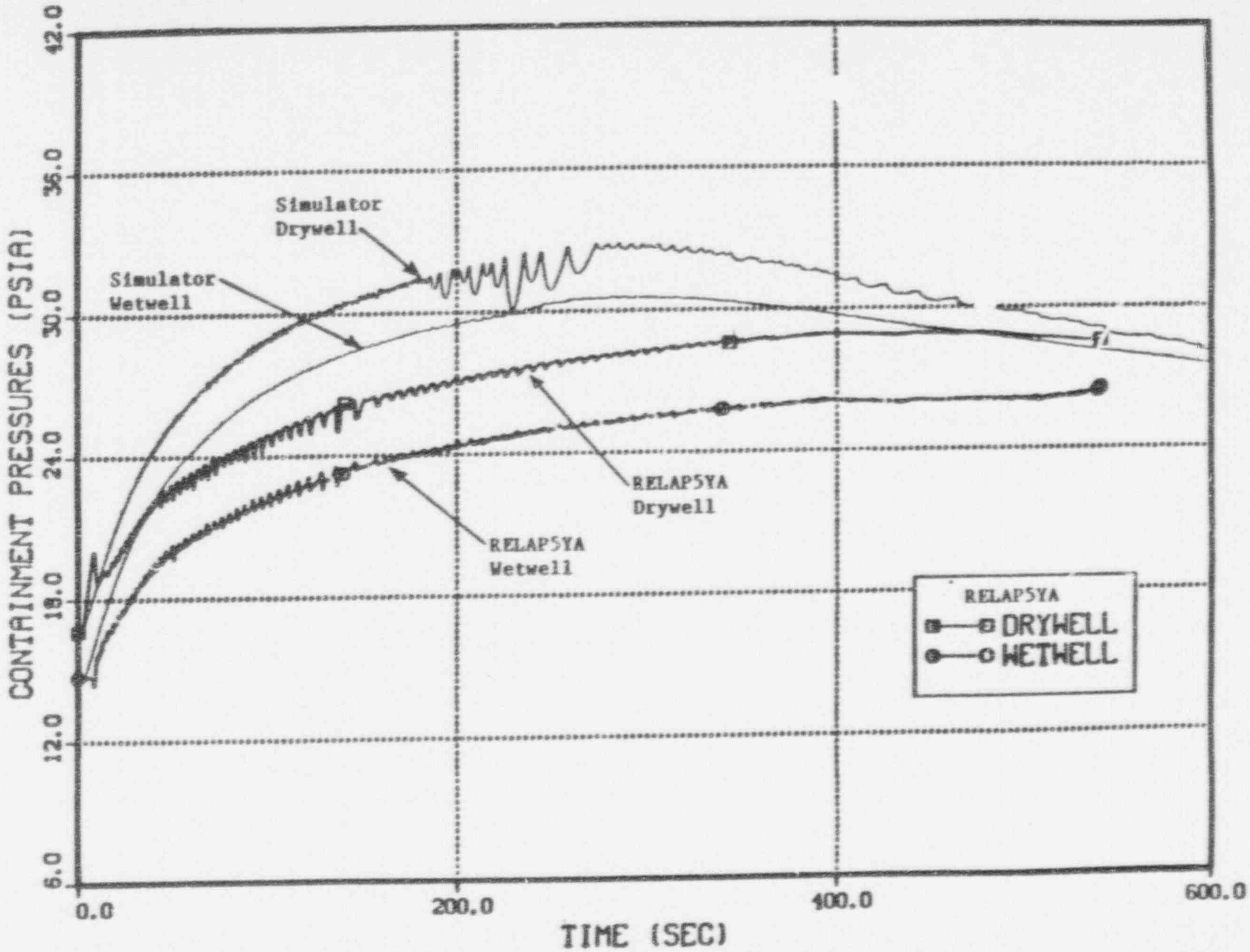


Figure 4-3.18 VY SB-LOCA Containment Pressures

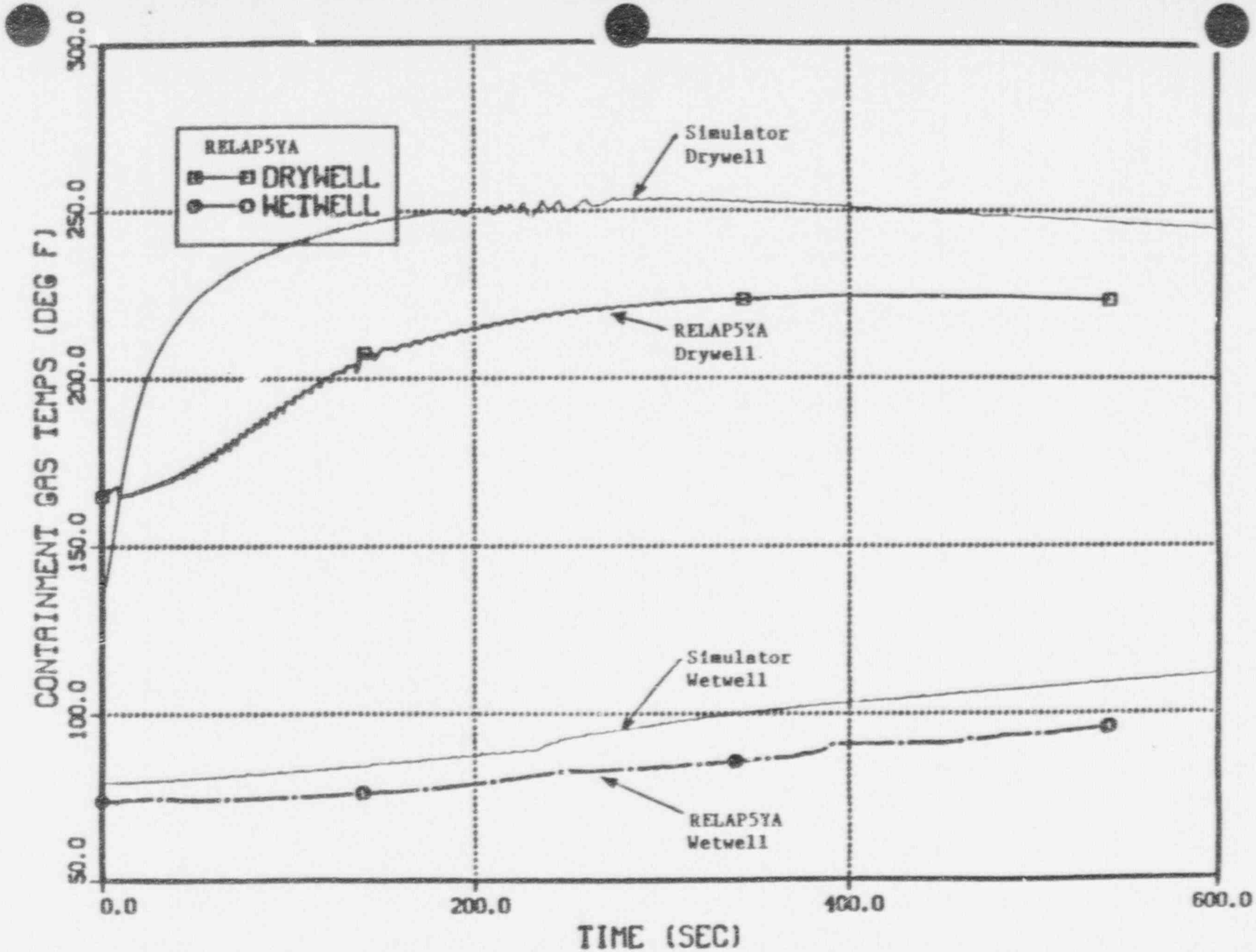


Figure 4-3.19 VY SB-LOCA Containment Gas Temperatures

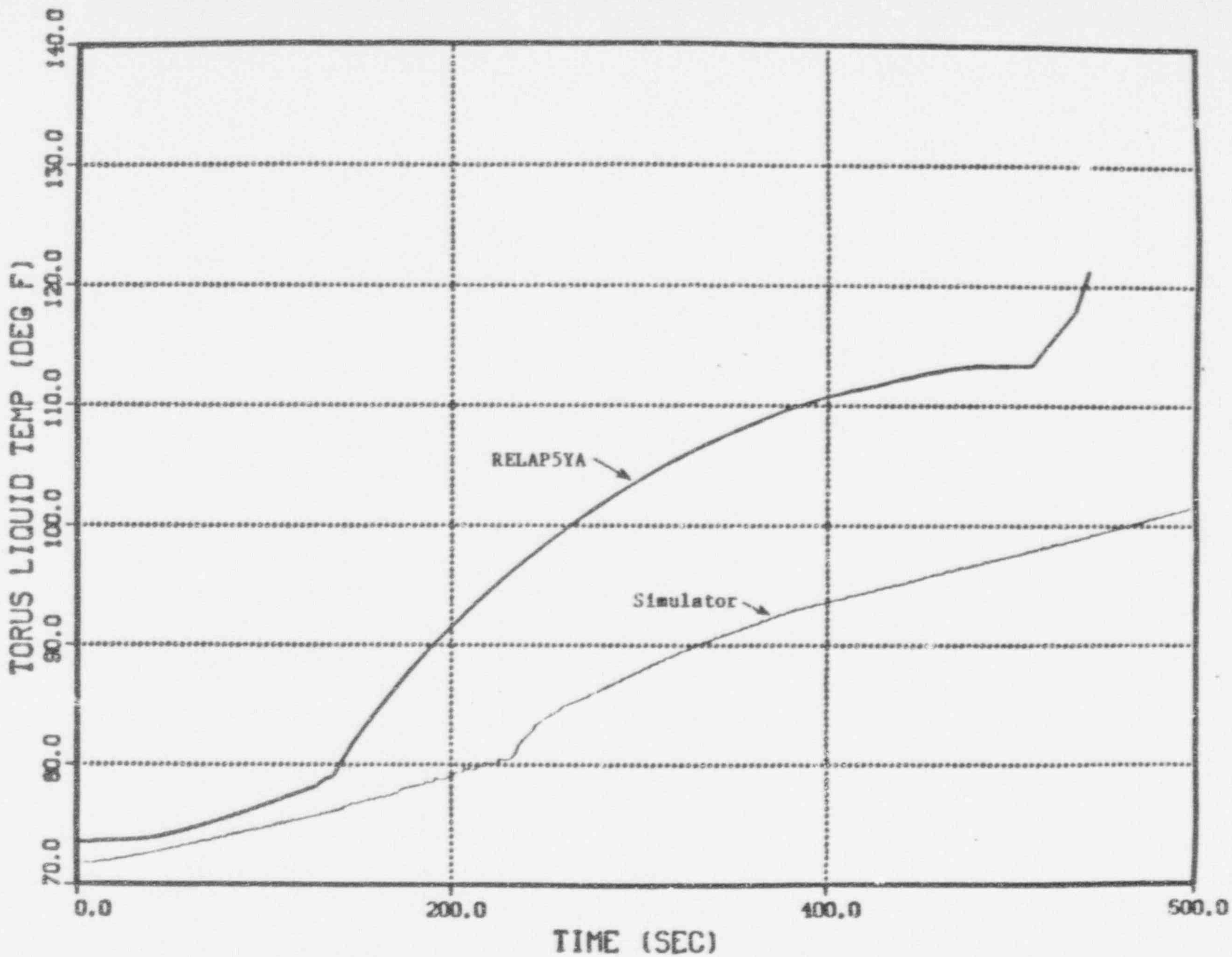


Figure 4-3.20 VY SB-LOCA Torus Liquid Temperature

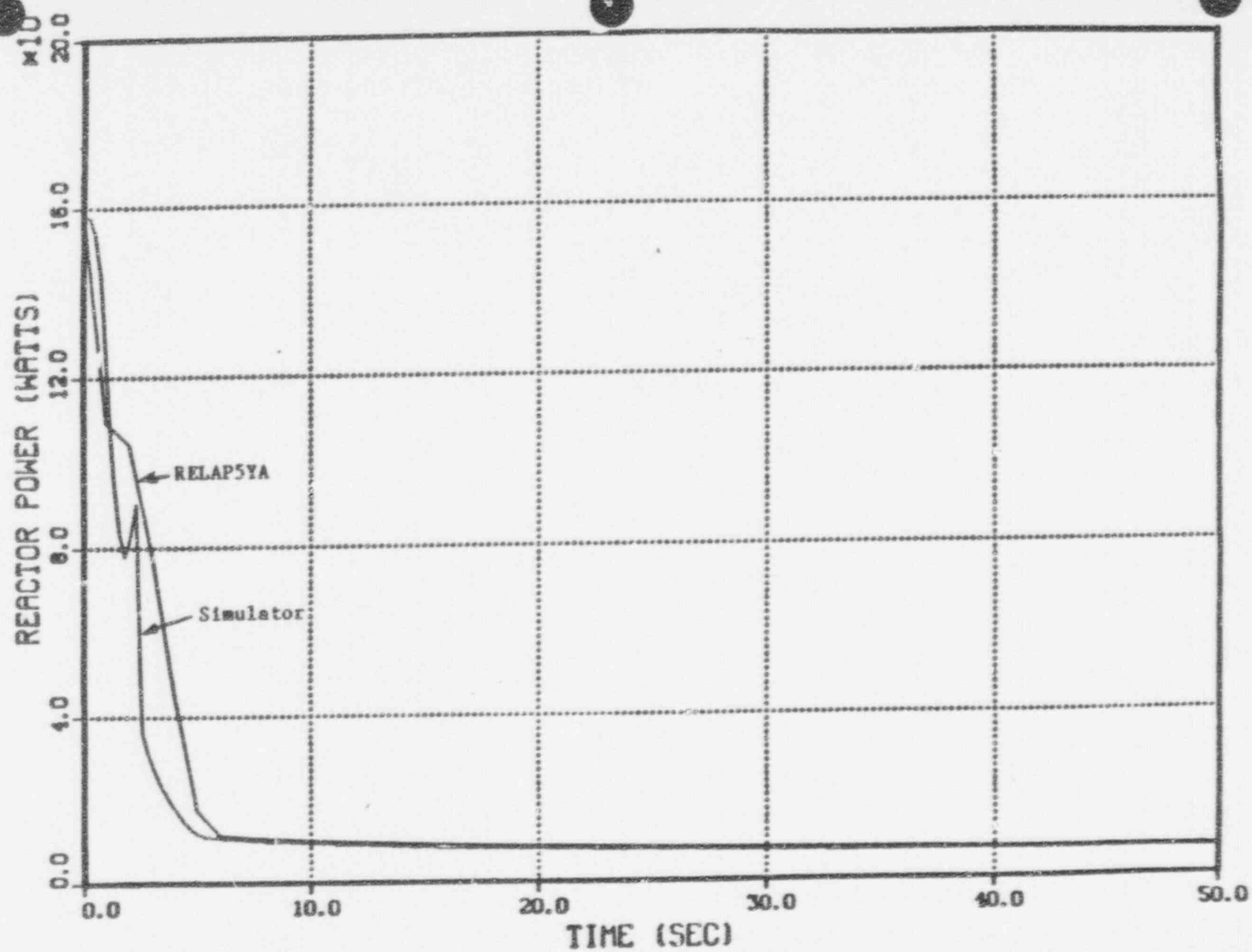


Figure 4-4.1 VY MSLB Core Thermal Power

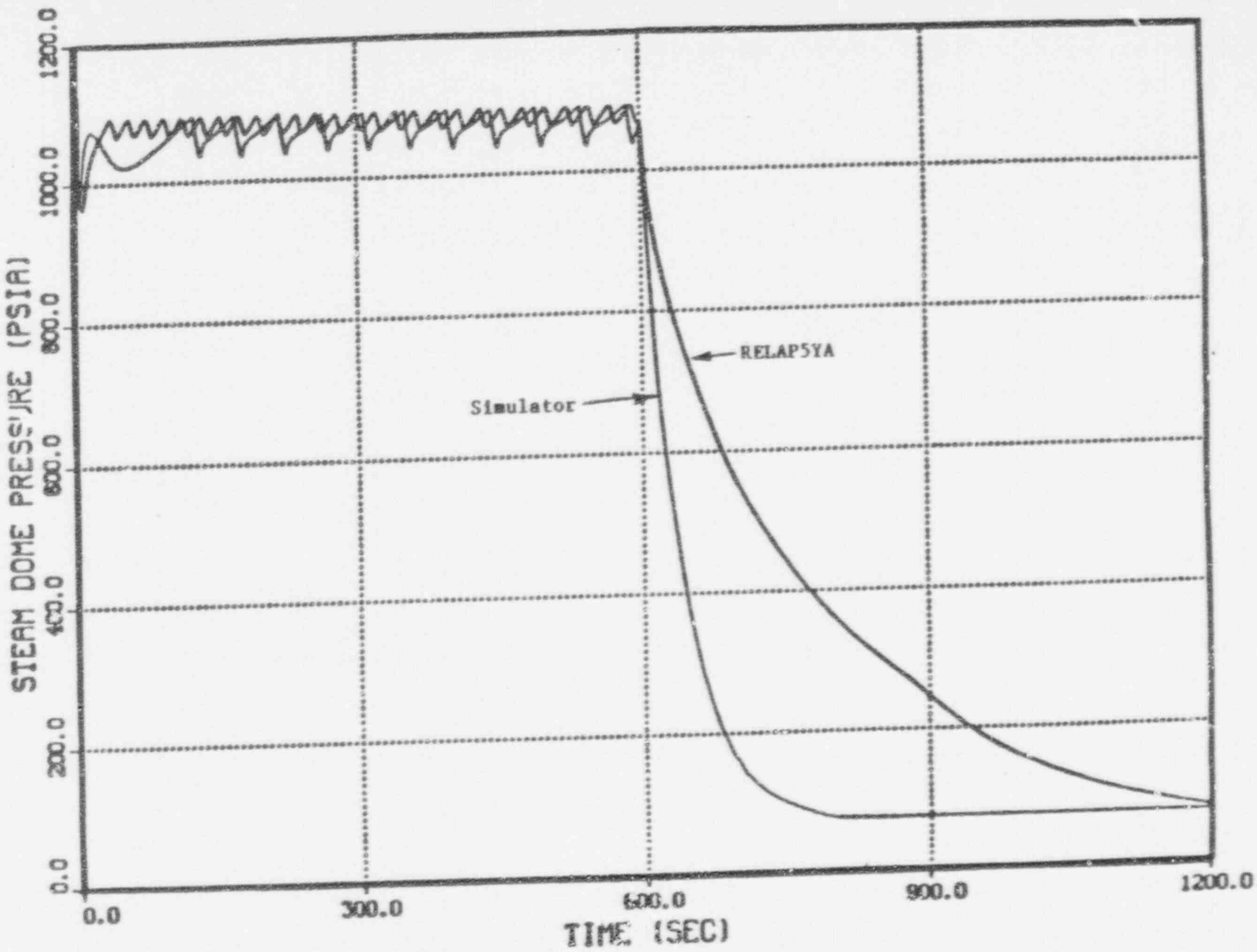


Figure 4-4.2 VY MSLB Reactor Vessel Steam Dome Pressure

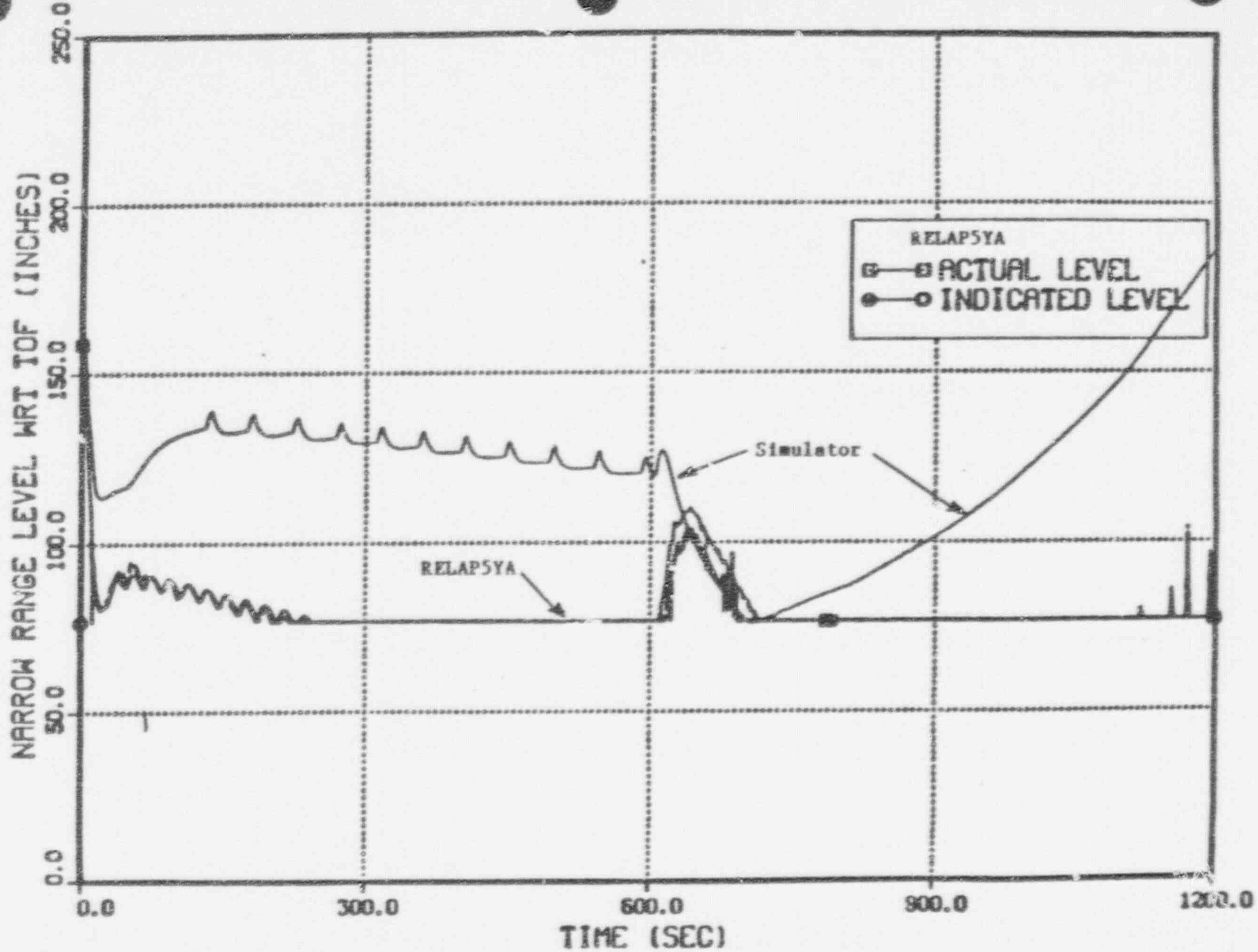


Figure 4-4.3 VY MSLB Reactor Vessel Narrow Range Water Level

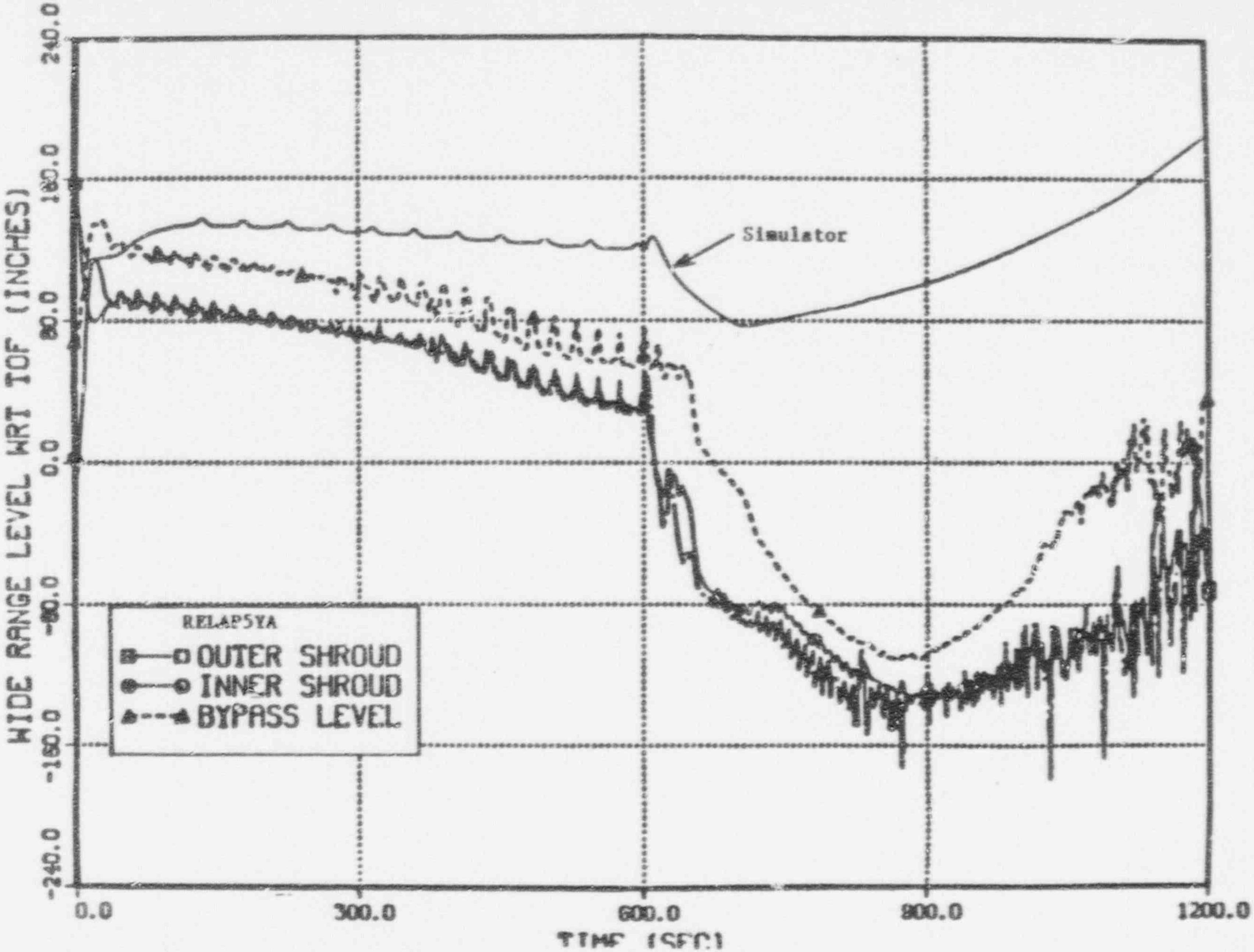


Figure 4-4.4 VY MSLB Reactor Vessel Wide Range Water Level

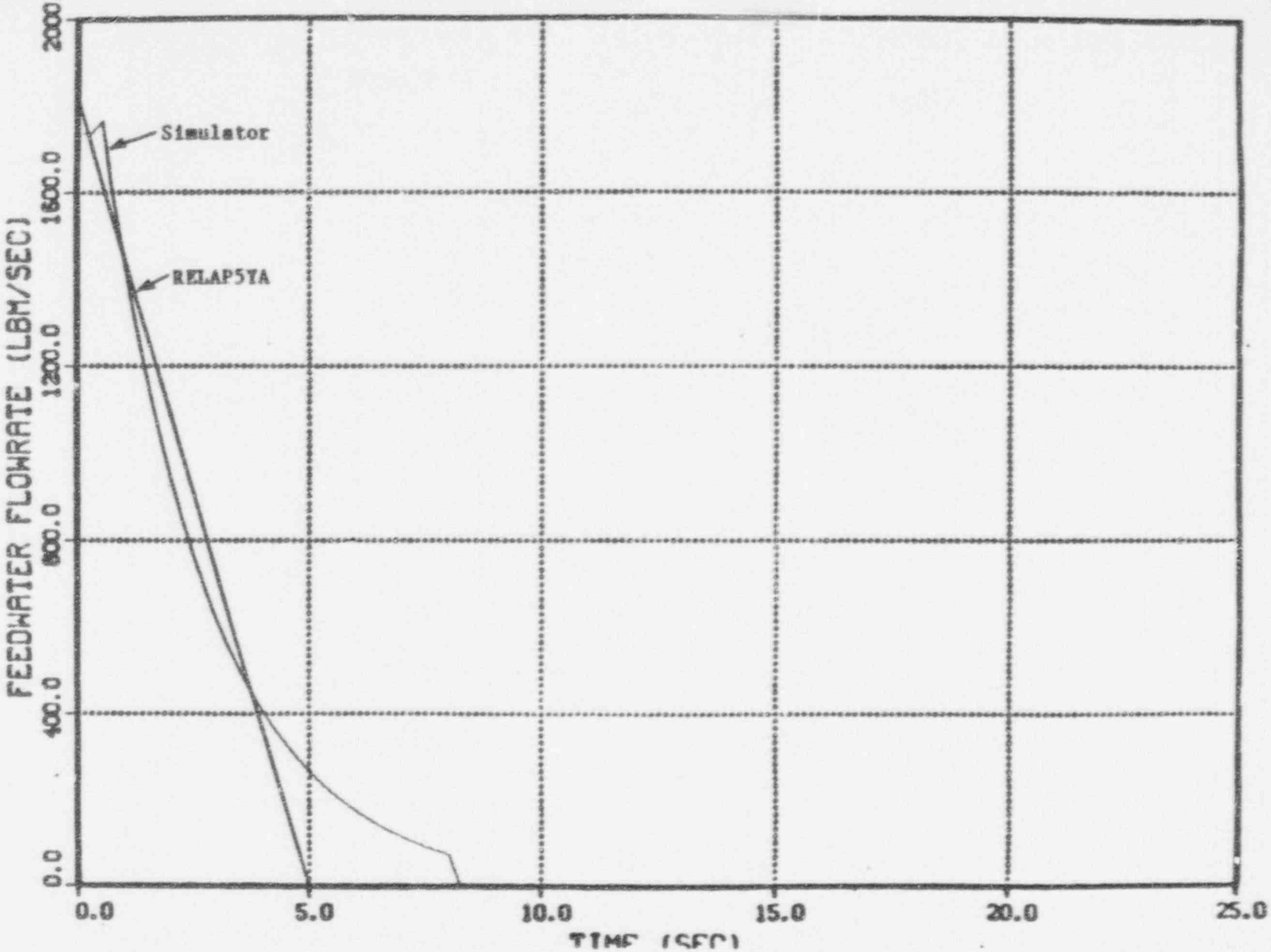


Figure 4-4.5 VY MSLB Feedwater Flow

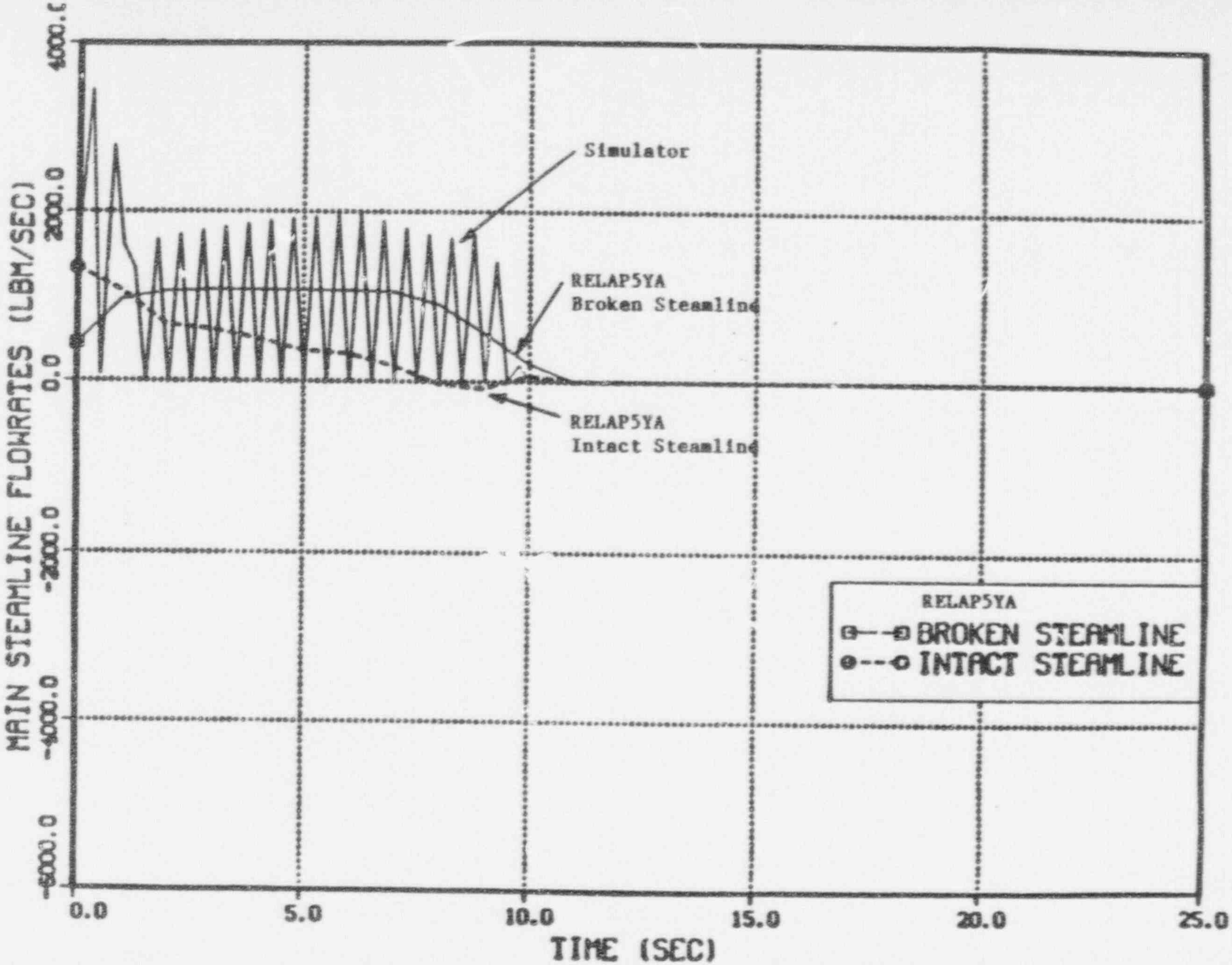


Figure 4-4.6 VY MSLB Steam Line Flow

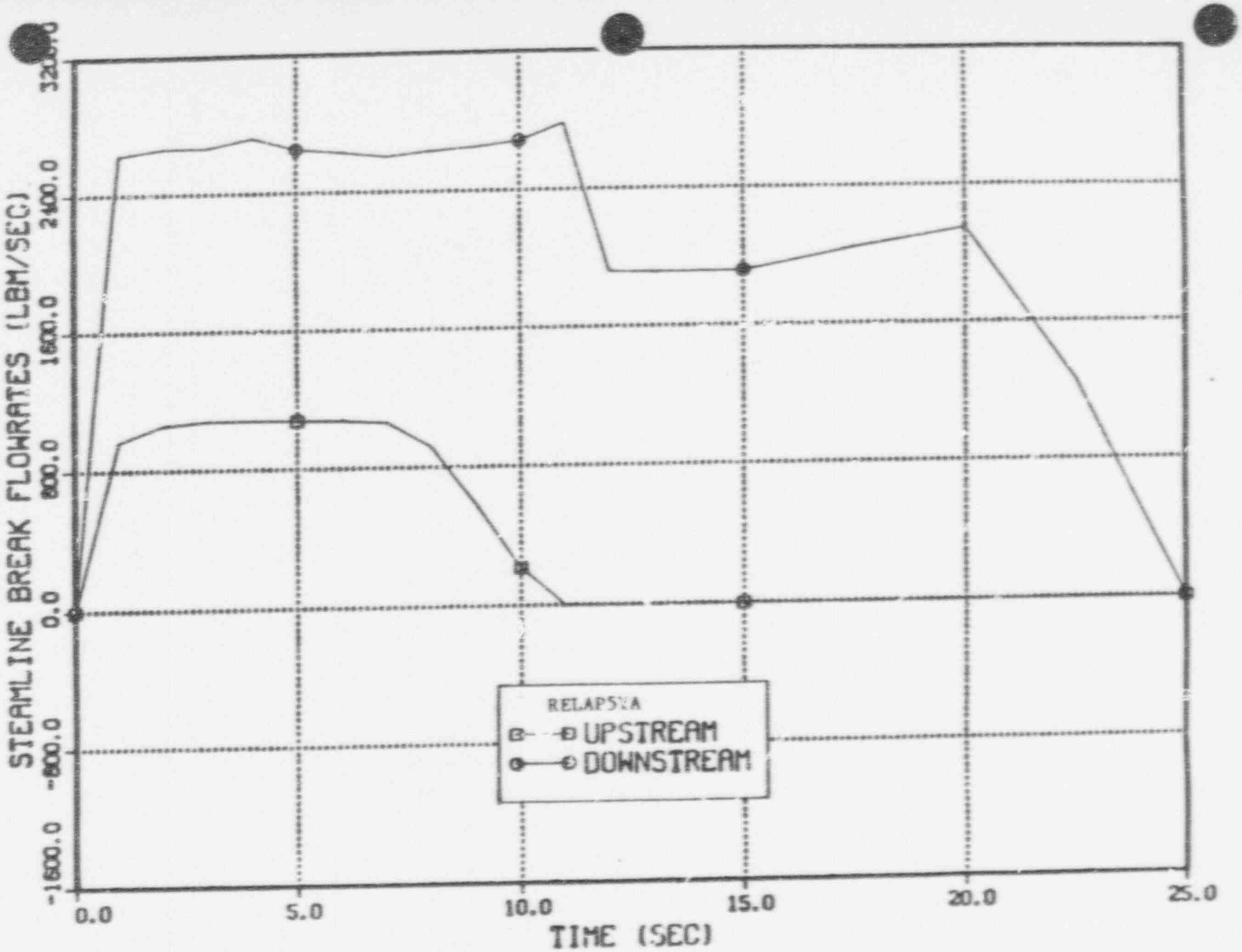


Figure 4-4.7 VY MSLB Steam Line Break Flow

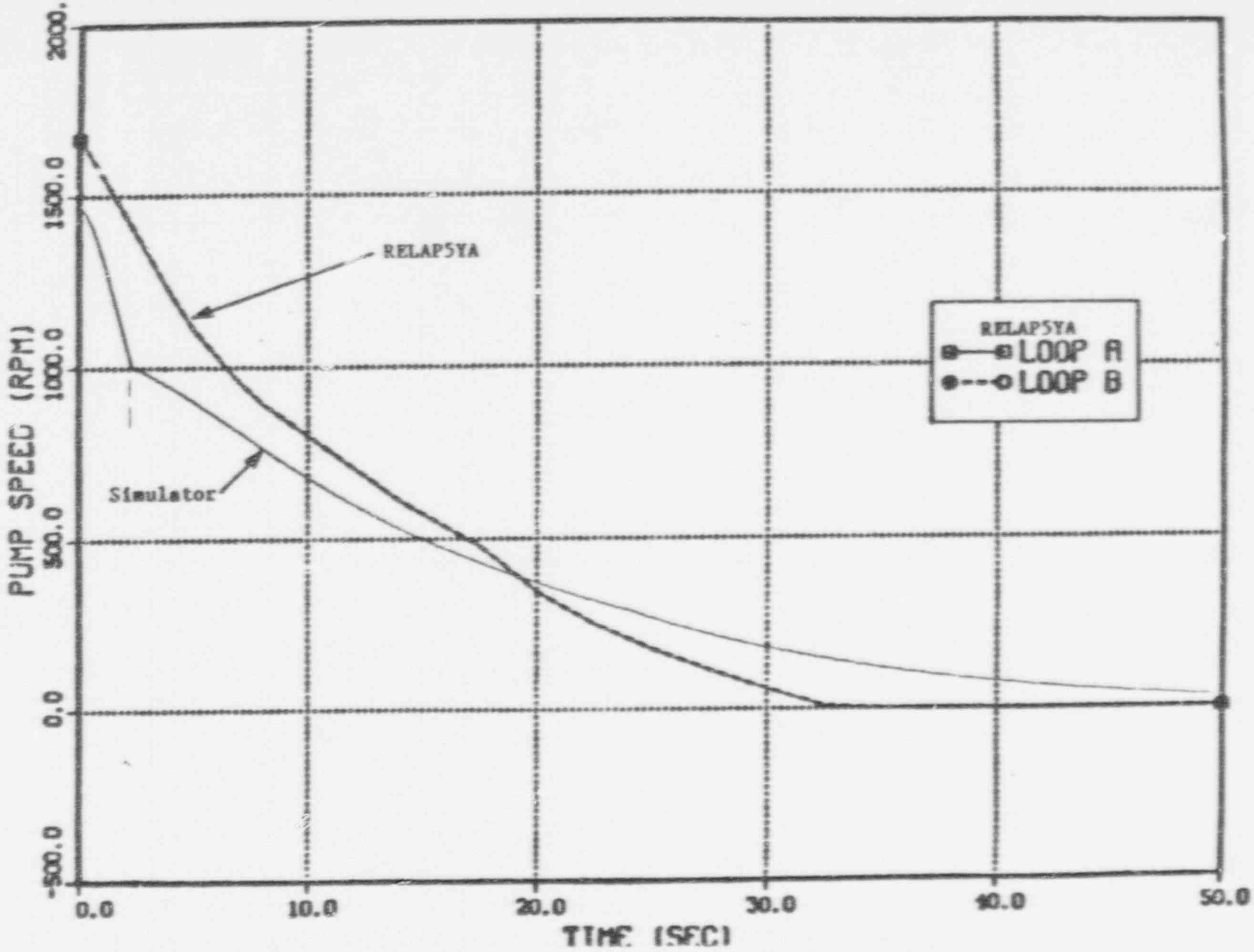


Figure 4-4.8 VY MSLB Reactor Recirculation Loop Pump Speeds

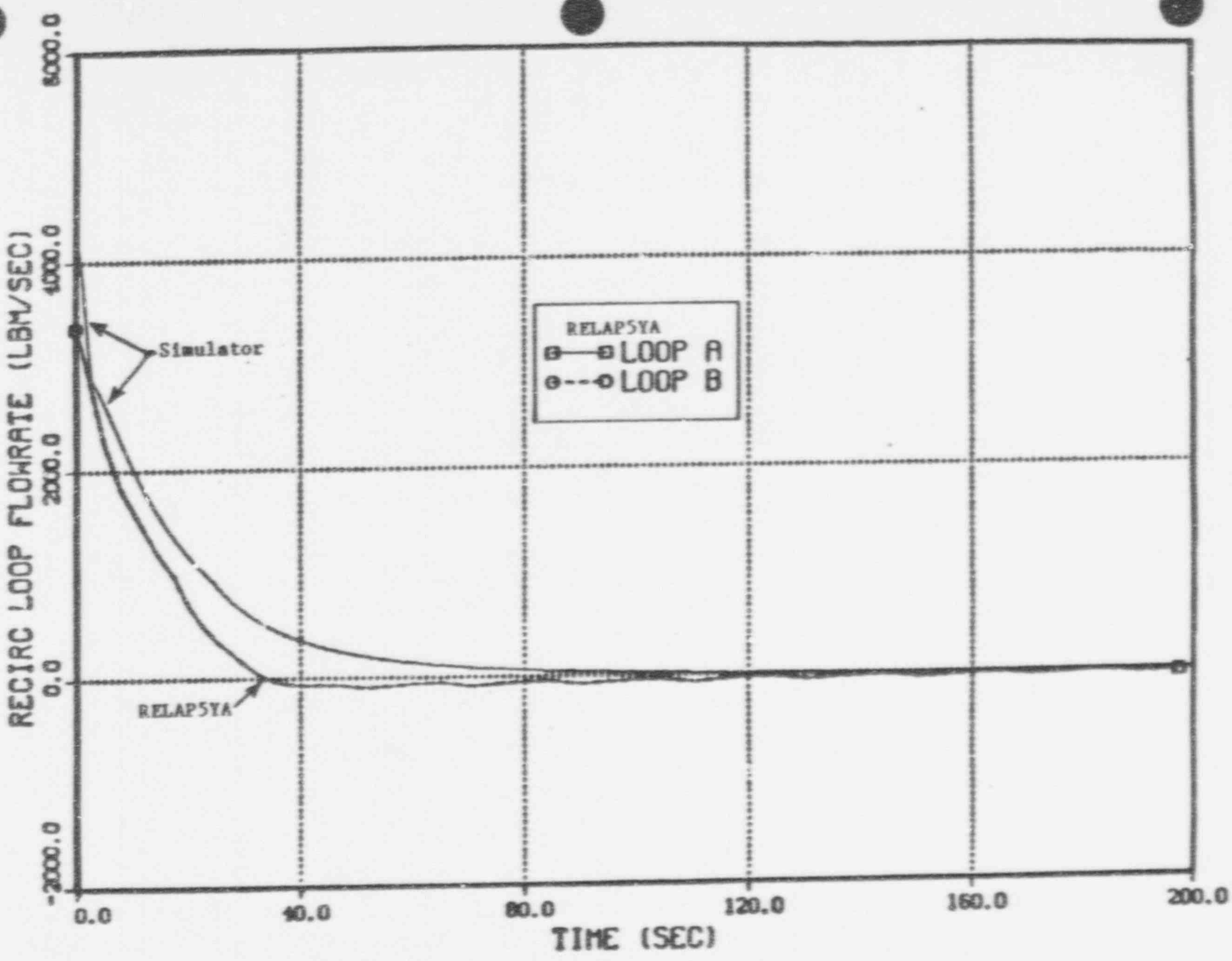


Figure 4-4.9 VY MS23 Reactor Recirculation Loop Flows

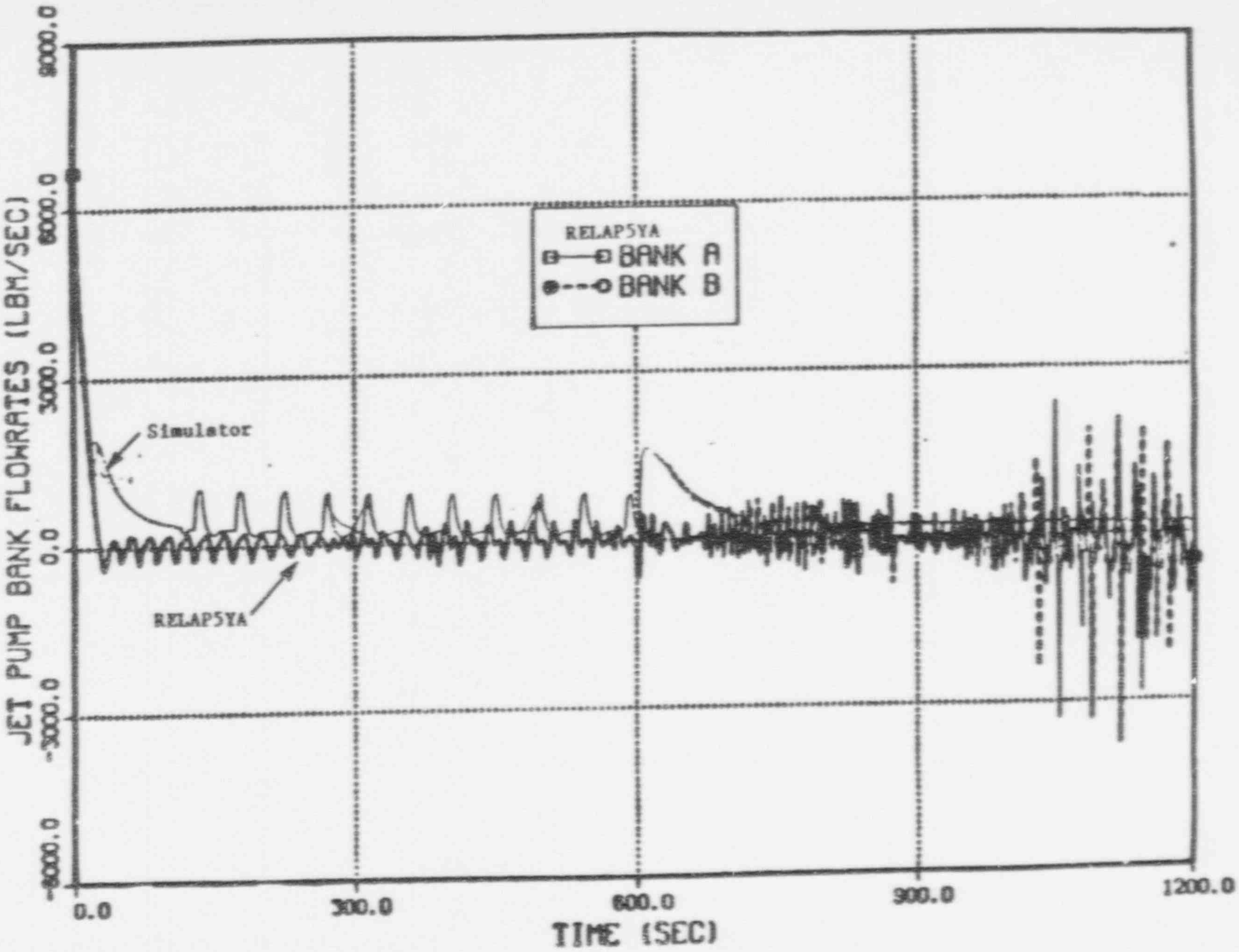


Figure 4-4.1d VT MSLB Jet Pump Flow

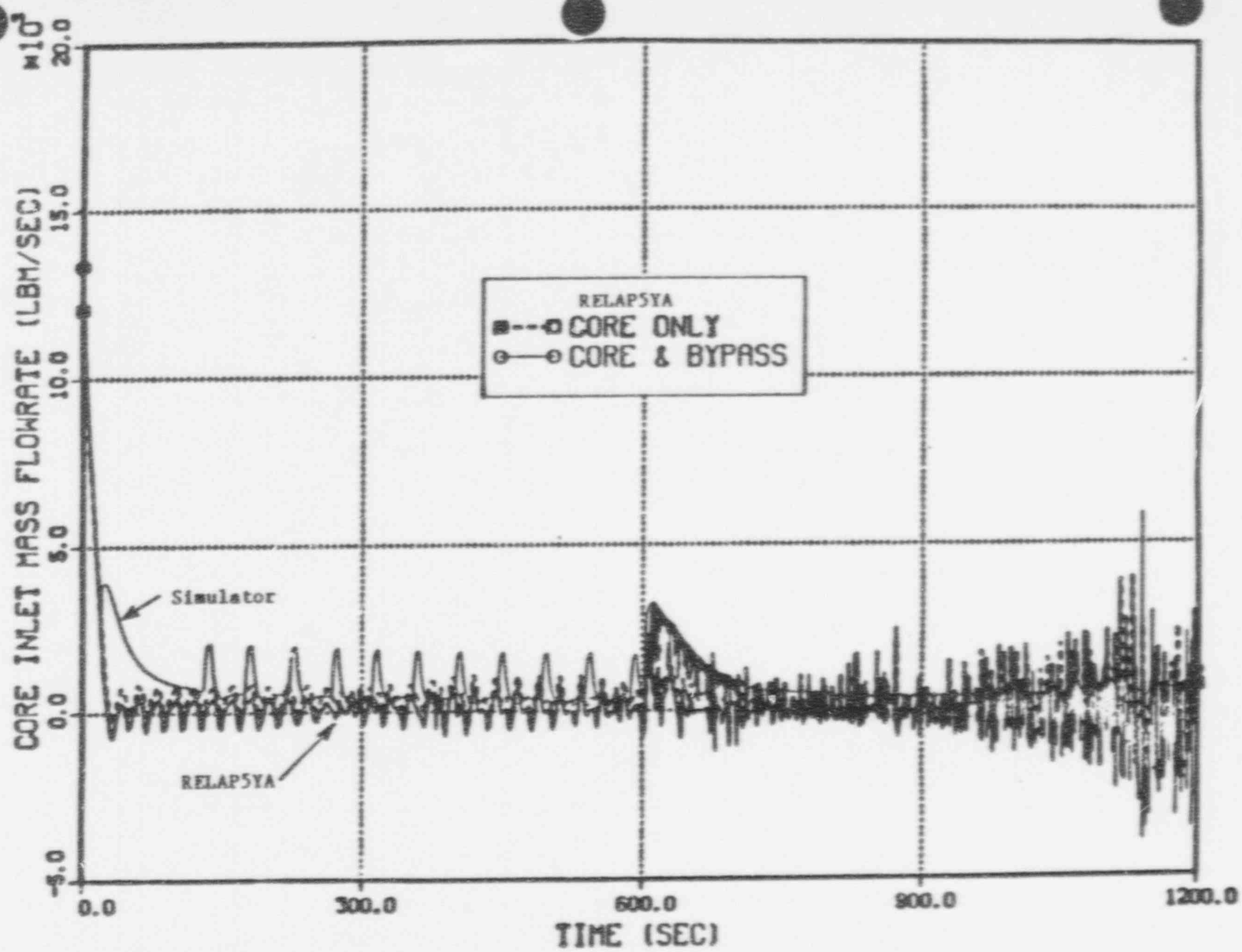


Figure 4-4.11 VY MSLB Core Inlet Flow

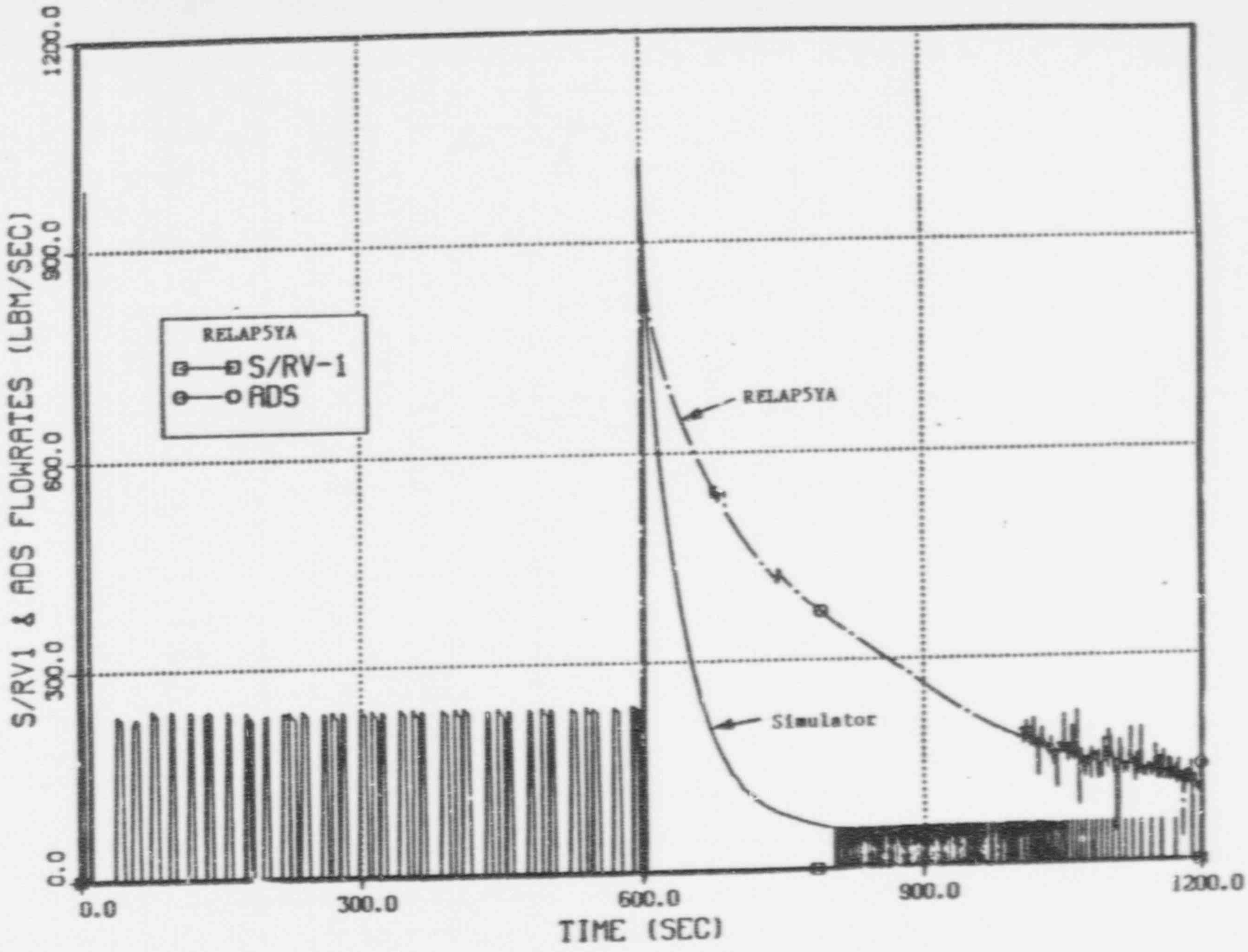


Figure 4-4.12 VT MSLB Safety/Relief Valve(s) Flow

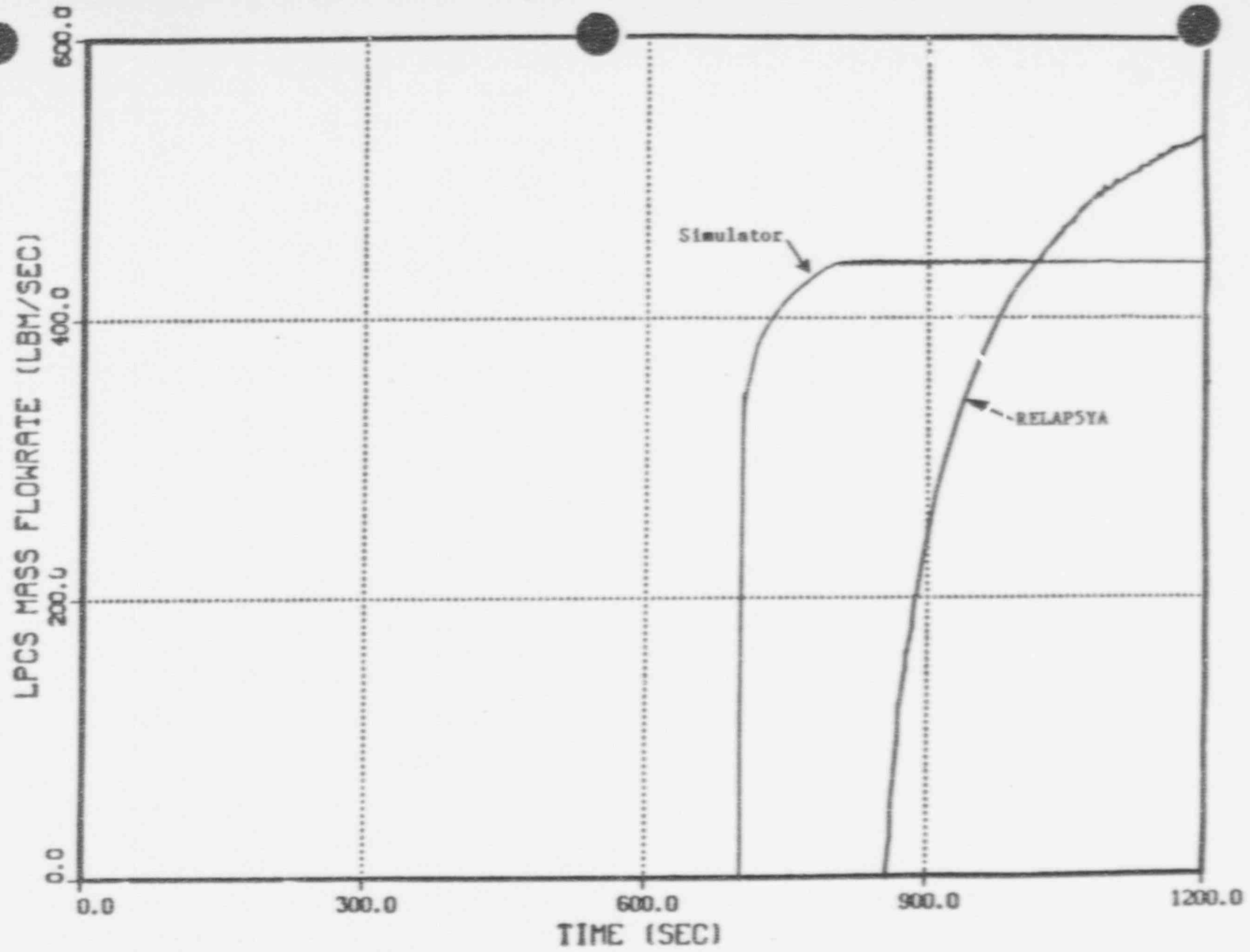


Figure 4-4.13 VT MSLB LPCS Mass Flow

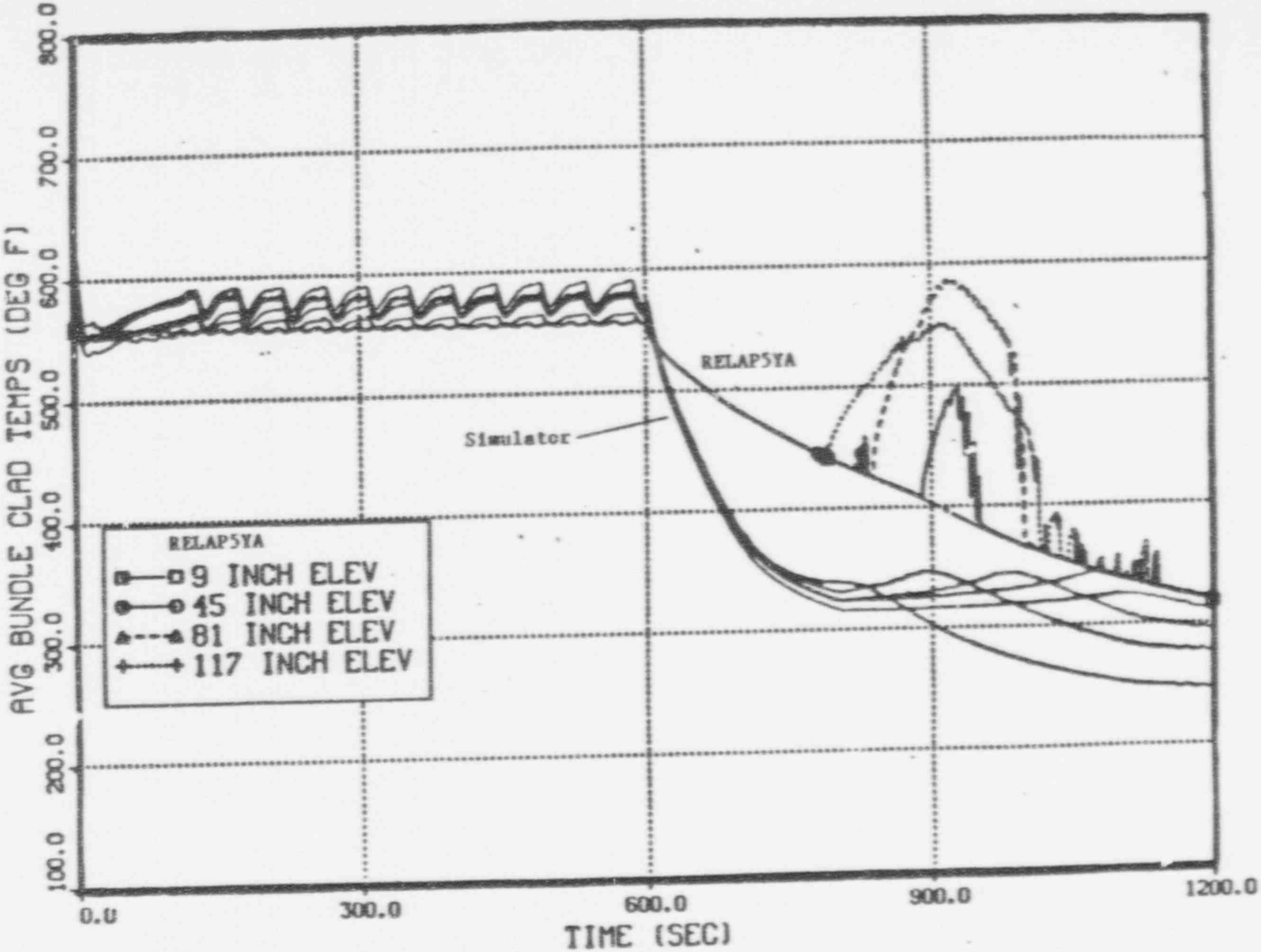


Figure 4-4.14 VY MSLB Average Bundle Clad Temperatures

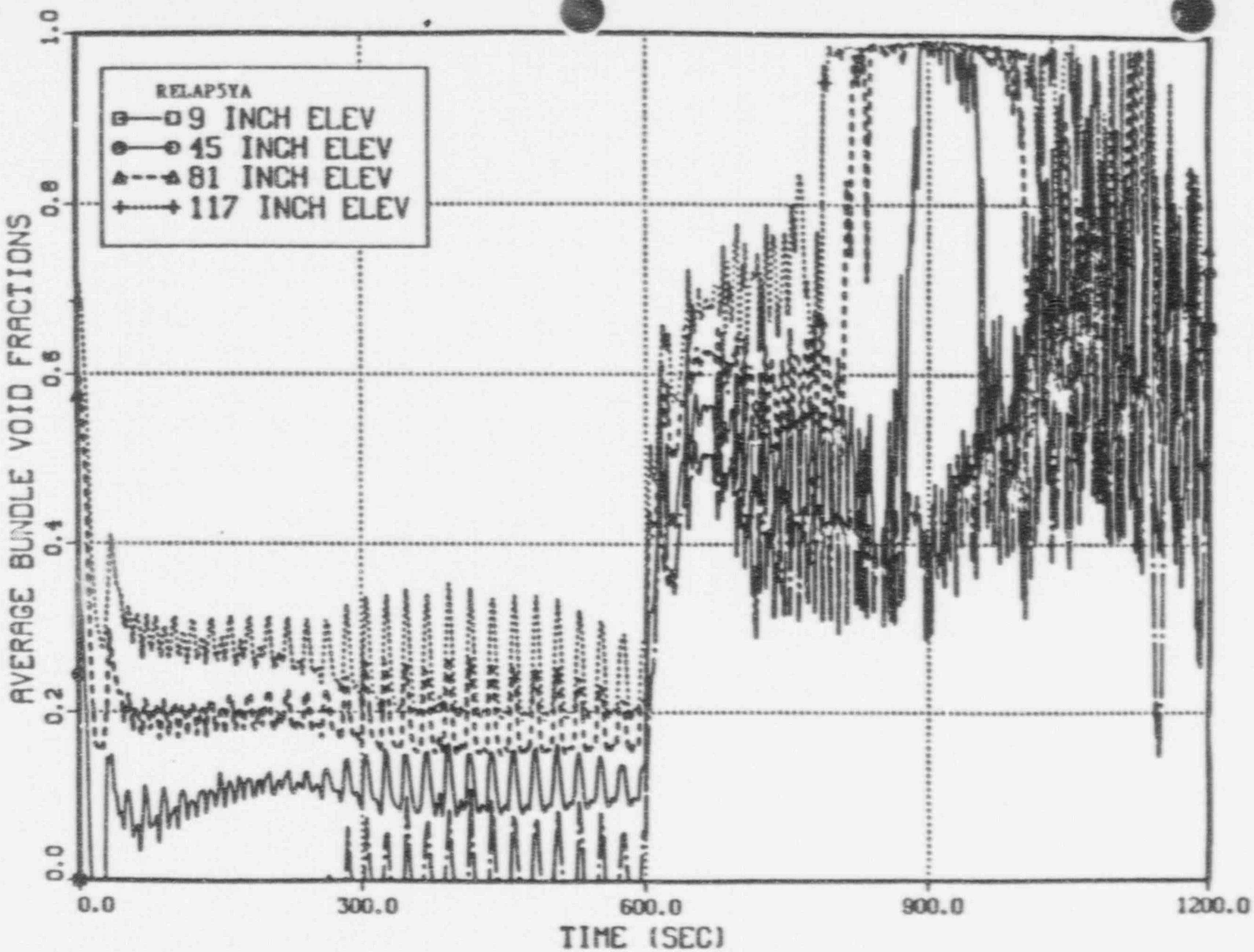


Figure 4-4.15 VT MSLB RELAP5YA Core Void Fraction

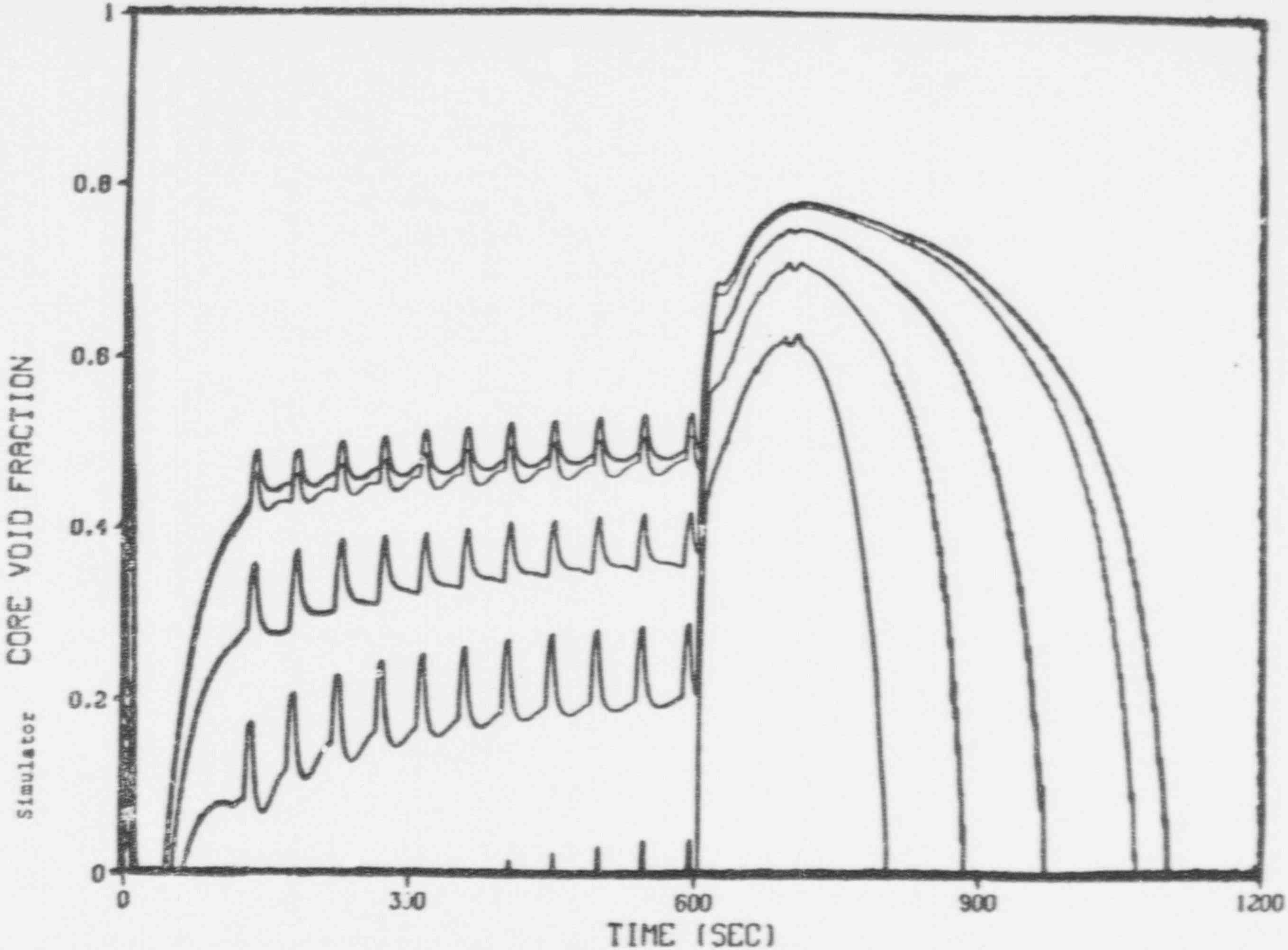


Figure 4-4.16 VY MSLB Simulator Core Void Fraction

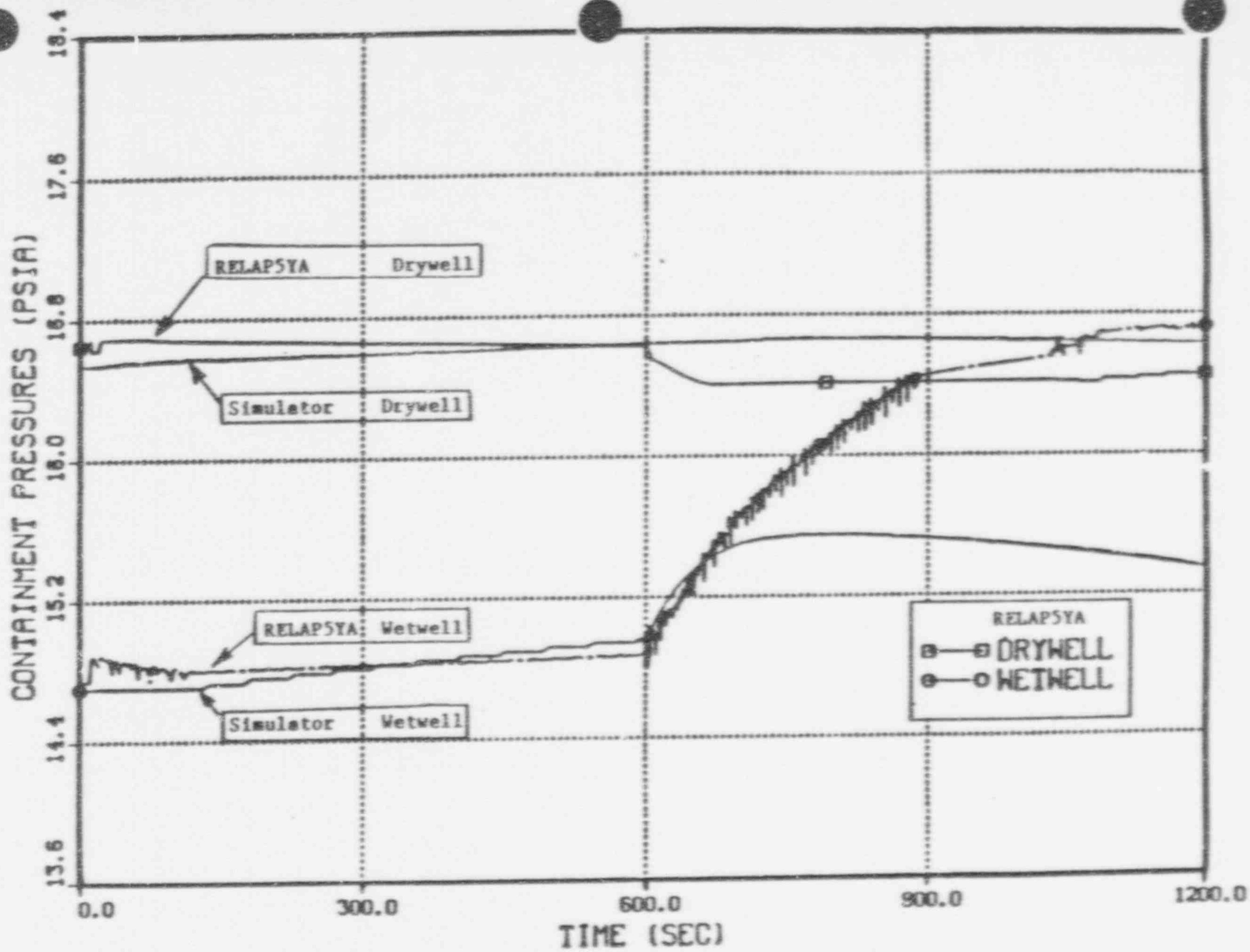


Figure 4-4.17 VY MSLB Containment Pressures

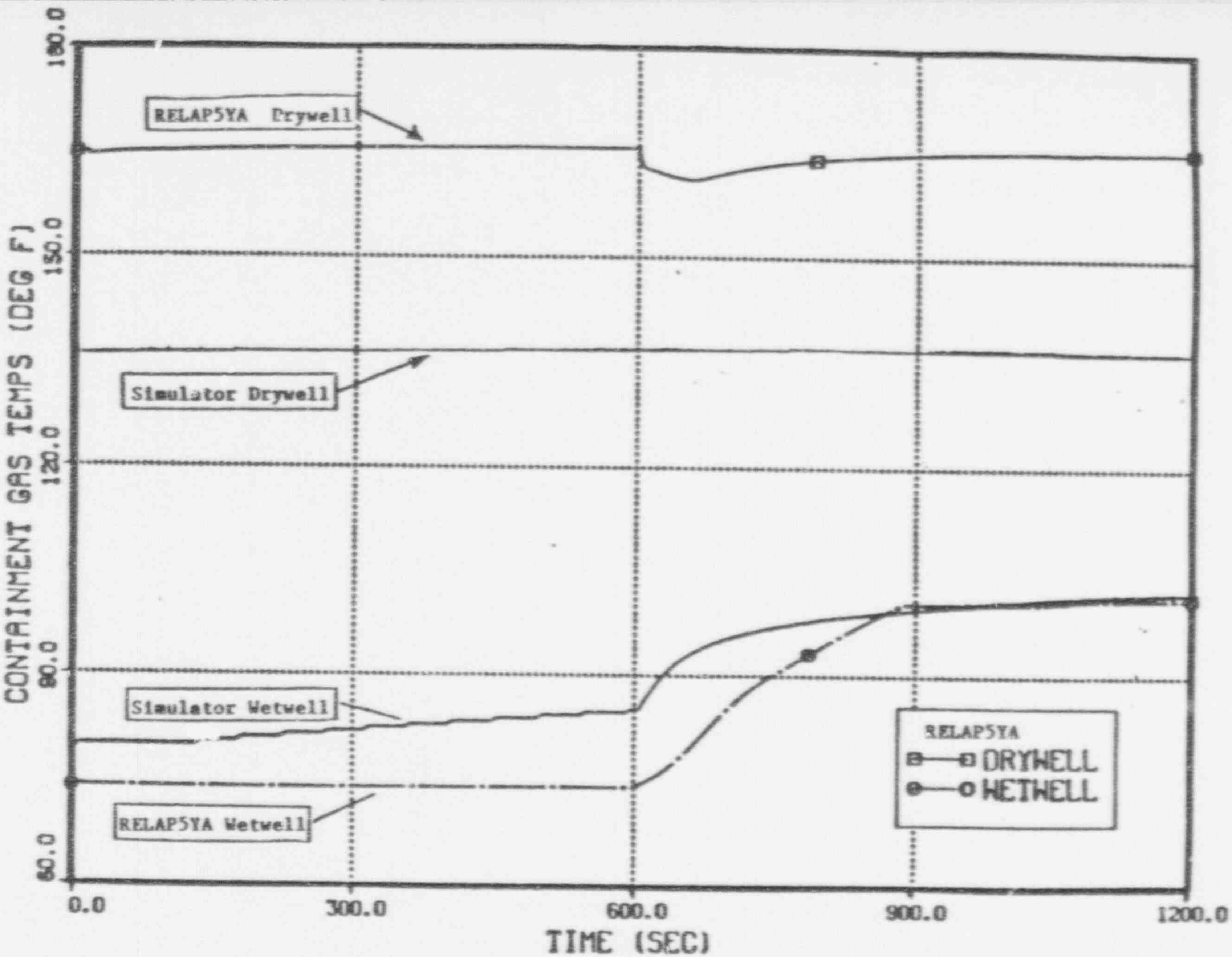


Figure 4-4.18 VT MSLB Containment Gas Temperature

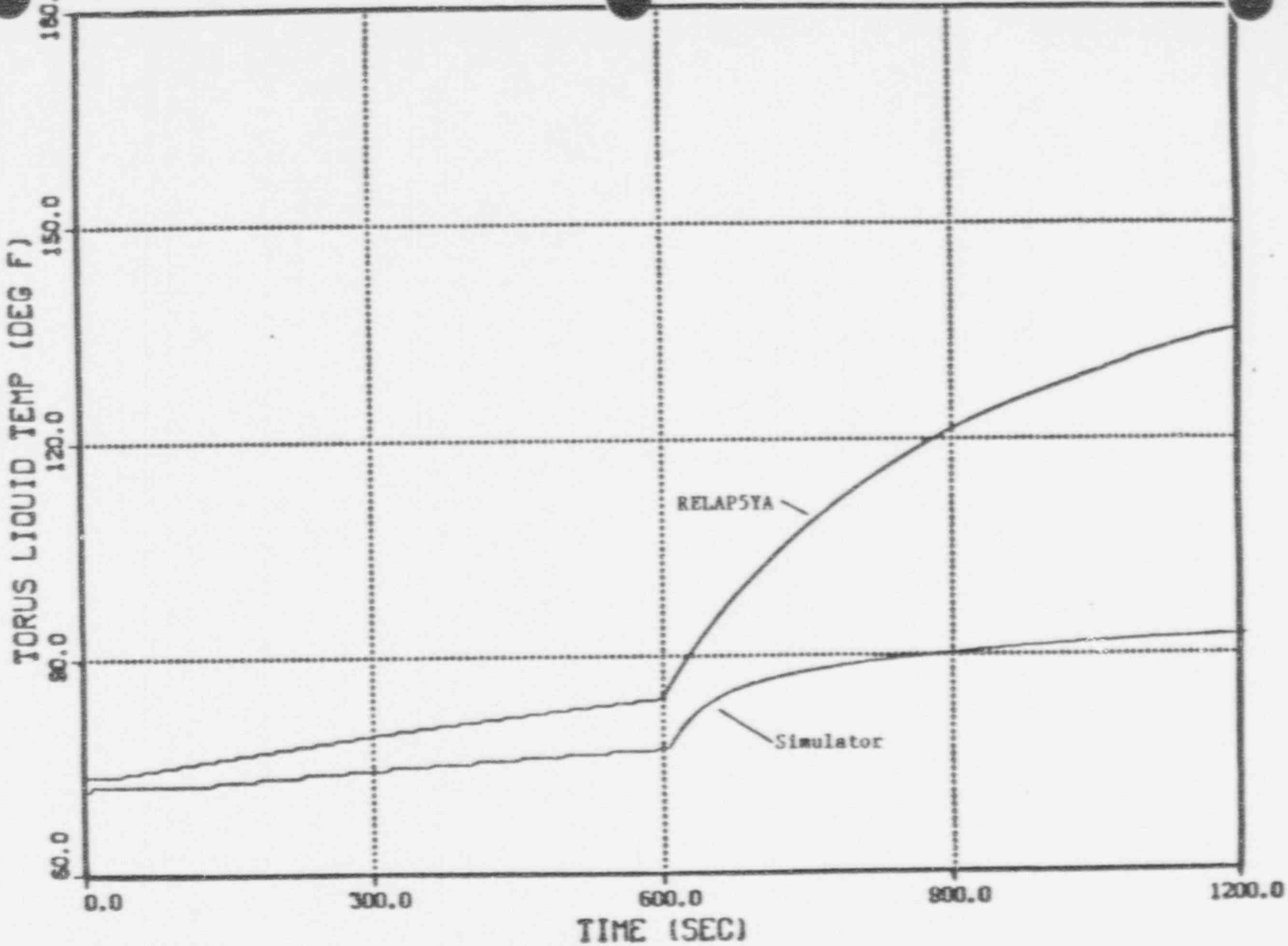


Figure 4-4.19 VY MSLB Torus Liquid Temperature

5.0 CONCLUSIONS

The Simulator provided an adequate representation of the plant response to the scenarios tested. The requirement for realtime computing resulted in several areas where modeling detail was sacrificed and impacted the results. Several areas were identified where we believe that further refinements to the model would produce more realistic simulations. These are discussed in the next section.

6.0 SUGGESTIONS FOR FUTURE IMPROVEMENTS

The following modifications would improve the Simulator's performance relative to the benchmark result. These suggestions are listed in the order that we judge would yield the most improvement in the Simulator's fidelity.

1. Reformulate the core model to allow explicit representation of the bypass region separately from the fuel assemblies. This would provide a more realistic simulation of the steady state and transient core region thermal-hydraulic and neutronic characteristics that include the following:

   a. fluid inventory and distribution
   b. core flow rate distribution
   c. core reactivity components

   This subsequently impacts the calculated reactor vessel water level and associated water level trips that occur during events where fluid from the downcomer region flows toward or from the core region.

2. Improve the models that represent the reactor vessel downcomer region in order to enhance the the simulation of the Reactor Water Level Measurement System. Many operator and automatic actions depend upon the accurate simulation of the vessel water level.

3. Account for liquid entrainment within steam, so that the flow rates and enthalpies exiting the ADS and pipe break locations will be represented more realistically. This would then yield more realistic depressurization rates and timing of significant events.

4. Properly model HPCI injection into the feedwater lines instead of direct injection into the core region. Also, correct the HPCI pumped fluid temperature to reflect the

source of the fluid ie. the condensate storage tank or the torus. These changes would improve the simulation of the fluid distribution within the vessel and the core thermal response for all events where HPCI is activated.

5. Properly model LPCI injection into the recirculation loops instead of direct injection into the core region. This change would improve the simulation of the fluid distribution within the recirculation loops, lower plenum and core, as well as the core thermal response for events where LPCI is injected.

6. Correct the Reactor Recirculation Loop M/G Set inertias and hydraulic characteristics. This would provide more realistic recirculation loop flow coastdown characteristics and improve the core thermal hydraulic response.

7. Correct the jet pump characteristics for off nornal conditions. This would provide a more realistic hydraulic response within the reactor vessel and core regions.

8. Modify the malfunctions used to inhibit complete control rod insertion to allow only 96" of insertion instead of the present value of 114". This would allow simulations of Partial SCRAM events.

7.0 REFERENCES

1-2.1 D. M. VerPlanck, "SIMULATE-2: A Nodal Core Analysis Program for Light Water Reactors," YAEC-1392P, dated June 1984

1-2.2 R. T. Fernandez, R. K. Sundaram, J. Ghaus, J. N. Loomis, L. Schor, R. C. Harvey, and R. Habert, "RELAP5YA - A Computer Program for Light-Water Reactor System Thermal-Hydraulic Analysis, Volume I: Code Description," YAEC-1300P, dated October 1982

1-2.3 J. H. McFadden et al, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Equations and Numerics," NP-1850-CCM, dated May 1981

2-1.1 Memorandum, T. G. Stetson to Cycle 9 File, "Control Rod Sequence Exchange @ Power, 9-A-1(2) to 9-B-2(2)," dated July 26, 1982

3-0.1 Memorandum, K. E. St.John to R. E. Sojka, "VY Simulator Initial Conditions Assumptions," NES 84-122, dated November 6, 1984.

3-0.2 Memorandum, K. E. St.John to M. J. Marian, "Schedule for VY Simulator Benchmarking," NES 86-7 dated January 29, 1986.

4.4-1 R. T. Fernandez, R. K. Sundaram, J. Ghaus, J. N. Loomis, L. Schor, R. C. Harvey, and R. Habert, "RELAP5YA - A Computer Program for Light-Water Reactor System Thermal-Hydraulic Analysis, Volume III: Code Assessment," YAEC-1300P, dated October 1982

4.4-2 L. Schor, "RELAP5YA Simulation of FSTF Containment Tests," to be issued

A-1.1 A. A. F. Ansari and J. T. Cronin, "Methods for the Analysis of Boiling Water Reactors a Systems Transient Analysis Model (RETRAN)," YAEC-1233, dated April 1981.

A-2.2 R. T. Fernandez and H. C. daSilva, "Vermont Yankee BWR Loss of Coolant Accident Analysis Method," YAEC-1547, dated June 1986

A-2.3 P. A. McGahan and M. A. Sironen, "Vermont Yankee Cycle 9 Summary Report," YAEC-1367, dated June 1983

A-2.4 M. A. Sironen and R. C. Potter, "Vermont Yankee Cycle 10 Summary Report," YAEC-1438, dated September 1984

A-2.5 R. C. Potter, "Vermont Yankee Cycle 11 Summary Report," YAEC-1513, dated February 1986

APPENDIX A

ENGINEERING MODEL DESCRIPTIONS

The engineering models described in the following sections are based on models employed in normal licensing activities within NED in support of VYNPS. The RETRAN-02 code is used by the Transient Analysis Group to evaluate potential operational transients used to define the operating MCPR limit. The RELAP5YA code was developed to provide LOCA licensing capability for use in analyzing various size potential Loss of Coolant Accidents at VYNPS. The SIMULATE-2 code is used by the Reactor Physics Group to calculate steady state and operational manuvers.

A.1 RETRAN Model Description

The VY ATWS RETRAN model was created from the VY RETRAN Reload model used in the VY Reload Licensing Analysis. It models the NSSS up to the Turbine Control Stop Valves, and consists of 33 volumes and 47 junctions, as well as Control Blocks for the Recirculation, Turbine and Feedwater control systems. A detailed description of the VY RETRAN Reload model is contained in Reference A-1.1. A simplified nodal diagram of the VY ATWS RETRAN model is shown in Figure A-1.1. Figures A-1.2, A-1.3 and A-1.4 depict the control logic for the Recirculation, Turbine and Feedwater control systems. Table A-1.1 contains important information pertaining to the VY ATWS RETRAN model. Table A-1.2 provides Safety/Relief Valve information.

A.2 RELAP5YA Model Description

A.2.1 RELAP5YA Computer Code

RELAP5YA (Reference A-2.1), a computer program for light-water reactor system thermal-hydraulic analysis, has been adapted by Yankee Atomic Electric Company for LOCA analyses. RELAP5YA provides a consistent, integral analysis capability of the system and core response to LOCA events and other plant transients. YAEC will use this program as a major part of its method to analyze the entire BWR break spectrum and the PWR small break spectrum in a manner that conforms to U.S. Nuclear Regulatory Commission requirements contained in 10CFR50.46 and Appendix K. YAEC has extensively applied this program for other conservative and realistic analyses of LOCA events and transients.

RELAP5YA has been developed from the RELAP5 MOD1 Cycle 18 code that was originally developed by EG&G Idaho, Inc., under USNRC sponsorship, and publicly released.

A.2.2 Nuclear Steam Supply System Model

This section describes the RELAP5YA Vermont Yankee NSSS input model used in the Simulator benchmarking. A detailed description of the model is provided in Reference A-2.2. It models the NSSS up to the Turbine Stop Valves, and consists of 136 volumes, 148 junctions, and 136 heat structures as well as control systems and trip logic.

A.2.2.1 Hydrodynamic Modeling

Figures A-2.1, A-2.2 and A-2.3 show the Vermont Yankee NSSS nodalization diagram for the Large Break LOCA, Small Break LOCA, and Main Steam Line Break models, respectively. Table A-2.1 identifies the corresponding regions or systems, and summarizes the number of volumes, junctions, and heat structures used in the Vermont Yankee NSSS model. Active volumes mean all volumes except Time-Dependent

Volumes (TMDPVOLs). Likewise, active junctions mean all junctions except Time-Dependent Junctions (TMDPJUNs). Active heat structures mean those heat structures that model axial segments of fuel rods. Passive heat structures are those that model reactor vessel and pipe walls, and reactor vessel internal structures. Table A-2.2 lists information for the Safety/Relief Valve(s).

A.2.2.2 Core Power

The total core power is determined by the point reactor kinetics model in RELAP5YA. The model accounts for moderator void, Doppler, and SCRAM reactivity effects. All core power is conservatively assumed to be generated in the fuel, i.e., none is deposited in moderator, cladding, or passive heat structures. This power is distributed according to the Nodal Power Factor (NPF) entered for each active heat structure that represents a portion of UO2 fuel. Each nodal power factor is the product of three terms:

$$NPF = F_b \times F_r \times F_z$$

where:

F_b = Core region bundle fraction.

F_r = Core region radial power factor.

F_z = Core region axial power factor.

SIMULATE computer code (References A.2-1, A.2-2, and A.2-3) data for Vermont Yankee Cycles 9, 10, and 11 were reviewed to determine conservative values for each term in order to achieve bounding cycle-independent values. A chopped cosine shape was assumed for the axial power shape. The following values were selected from this review:

Core Region	F_b	F_r	F_z
Peripheral Low Power	0.315	0.585	1.40
Central Average Power	0.674	1.186	1.40
Central High Power	0.011	1.500	1.53

This yields a maximum local peaking factor in the high power bundles of 2.295 (1.50 x 1.53). This value is 11.4% larger than the highest value found during the review. This peak occurs in axial Node 5 located between 72 and 90 inches above the bottom of the fuel zone. Finally, the power within each fuel node is distributed according to flux depression factors obtained from FROSSTEY computer code results.

A.2.2.3 ECCS Modeling

The Emergency Core Cooling System at VY includes the following systems: HPCI, LPCI, LPCS, and ADS.

The HPCI system consists of one high pressure steam turbine assembly and a constant-flow pump assembly with associated piping, valves, controls, and instrumentation. This system is capable of delivering 4250 gpm over a broad range of vessel pressures (1,120 to 75 psid vessel to containment). The steam supply to the HPCI turbine is modeled with TMDPJUN 561. The HPCI ECCS flow is modeled using TMDPJUN 701. The flow versus time values in the RELAP5YA table account for a 20.33-second startup time followed by a 5-second ramp to 587 lbm/sec (100°F water) after the HPCI initiation signal occurs.

The two independent core spray systems are modeled as one combined system by TMDPJUN 721. A control variable (CV721) monitors the upper plenum to wetwell pressure difference (P206-P720). This parameter is used as the independent variable in the TMDPJUN table to determine the LPCS flow after the injection valve has opened. The table's values have been conservatively selected from actual VYNPS pump data. Specifically, the lower of the two-pump performance curves was used and the flow rates were reduced by 3% to allow for the estimated measurement uncertainties.

Each independent LPCI System is modeled separately since they inject into different recirculation loops. These are modeled by TMDPJUNs 741 and 761. Each has an associated control variable that monitors the injection pipe to wetwell pressure difference:

CV741 = P742 - P740

CV761 = P762 - P760

Each CV parameter is then used as the independent variable for the corresponding TMDPJUN table. The two tables are identical. The flows assume that two pumps are operating in each system and have been reduced by 150 gpm to allow for the estimated uncertainties.

The ADS is modeled as a trip valve (Junction 559) using the combined nozzle flow areas of the four safety/relief valves. When the ADS opens, the corresponding SR/Vs (if open) are quickly ramped closed. A two-phase discharge coefficient of 0.848 is used for these valves in order to better their rated conditions.

A.2.3 Containment Model

Vermont Yankee has a Mark I containment which consists of a drywell which surrounds the reactor vessel, 8 vent pipes connecting the drywell to the wetwell, and the wetwell with 96 downcomers. The containment also has a set of vacuum breakers from the torus to the drywell.

Figure A-2.3 shows the VY containment model which contains 15 volumes, 16 junctions, and 16 heat structures. The 15 volumes represent the drywell, vent system, wetwell, and vacuum breaker lines. The 16 heat structures represent the drywell wall, vent system walls, wetwell wall, reactor shield and reactor pedestal.

The containment model was coupled with the NSSS model through the hydraulic and instrumentation response in these regions in order to provide an integrated plant response to the accidents analyzed. Table A-2.3 lists the containment initial conditions.

A.3 SIMULATE Computer Code

SIMULATE-2 is a computer program employed in three dimensional nodal analysis of light water reactor power distributions. The program has been employed in studies of cycle length, power distributions, control rod patterns, and Xenon transients.

The reactor core is represented in SIMULATE-2 using the nodalization familiar in typical nodal analysis programs. For Vermont Yankee, one node is used per assembly in the horizontal plane and twenty-four are used vertically. The neutron balance equation is basically a one-group method: however, with two-group cross section input, a thermal leakage correction is calculated internally for each node to provide an approxiamte two-group result. The nodal coupling probabailities needed in the balance equation are calculated internally. These nodal intercation probabilities should provide satisfactory results in most cases without normalization. The calculations are executed only in the fueled area of the core with albedoes being used to terminate the neutron balnace equation at the core-reflector interfaces.

TABLE A-1.1

VY ATWS RETRAN Model Information

Component Description	Volume Number	Physical Volume (ft^3)	Flow Area (ft^2)	Bottom Elev (in)	Top Elev (in)
Pressure Vessel - See Figure A-1.1					
Lower Plenum	9	2185.05	126.362	0.000	207.5
Core Bypass Region	11	887.750	71.0200	207.5	357.5
Core Fuel Region	201-212	42.1262-Ea	40.0400-Ea	207.5	357.5
Upper Plenum	1	672.900	116.600	357.5	426.8
Standpipes*	2	137.090	25.8800	426.8	480.0
Separators*	3	179.200	48.5000	(392.5)	554.0
Steam Dome and Dryers*	22	(5322.86)	184.000	(382.5)	757.5
Downcomer	4	(1694.84)	85.0000	102.5	(382.5)
Recirculation Loop A - See Figure A-1.1					
Pump Suction Line	7	216.630	3.90000	-319.0	150.0
Recirculation Pump	13	40.0600	1.00E+6	-319.0	-267.0
Pump Discharge Line	8	315.730	2.31000	-216.0	303.4
Ten Jet Pumps	16	107.250	6.40000	98.00	299.0
Recirculation Loop B - See Figure A-1.1					
Pump Suction Line	5	216.630	3.90000	-319.0	150.0
Recirculation Pump	12	40.0600	1.00E+6	-319.0	-267.0
Pump Discharge Line	6	315.730	2.31000	-267.0	303.4
Ten Jet Pumps	15	107.250	6.40000	98.00	299.0
Steam Line - See Figure A-1.1					
Steam Piping	50	393.000	5.67300	43.00	595.0
Steam Piping	51	103.960	5.67300	-161.0	43.00
Steam Piping	52	432.516	5.67300	-161.0	29.00
Steam Piping	53	395.351	5.67300	-39.60	-18.60
Steam Piping	54	389.139	5.67300	-50.40	-29.40
Steam Piping	55	410.195	5.67300	-90.00	129.6

Control Systems

Turbine Control System — See Figure A-1.2

Feedwater Control System — See Figure A-1.3

Recirculation Control System — See Figure A-1.4

* Information bounded by parentheses were incorporated specifically for the VY ATWS RETRAN model.

TABLE A-1.1 (Continued)

VY ATWS RETRAN MODEL INFORMATION

Component Description	Number	Junction Area (ft^2)	Elevation (in)
Pressure Vessel - See Figure A-1.1			
Lower Plenum to Upper Plenum (Fuel)	200-212	40.4400-Ea	207.5-357.5
Lower Plenum to Core Bypass Region	16	0.31800	207.5
Core Bypass Region To Upper Plenum	18	44.9710	357.5
Upper Plenum to Standpipes	1	25.8770	426.8
Standpipes to Separators	2	25.8770	480.0
Separators to Steam Dome (Carry-over)	3	36.9800	554.0
* Separators to Steam Dome (Carry-under)	5	66.8000	(392.5)
* Steam Dome to Downcomer	21	114.800	(382.5)
Recirculation Loop A - See Figure A-1.1			
Downcomer to Pump Suction Line	1	3.90300	150.0
Pump Suction Line to Recirculation Pump	11	3.90300	-267.0
Recirculation Pump to Pump Discharge	12	3.90300	-267.0
Pump Discharge to Ten Jet Pumps (Force)	13	0.52720	299.0
Downcomer to Ten Jet Pumps (Suction)	14	1.48200	299.0
Ten Jet Pumps to Lower Plenum	20	11.0800	98.00
Recirculation Loop B - See Figure A-1.1			
Downcomer to Pump Suction Line	6	3.90300	150.0
Pump Suction Line to Recirculation Pump	7	3.90300	-267.0
Recirculation Pump to Pump Discharge	8	3.90300	-267.0
Pump Discharge to Ten Jet Pumps (Force)	9	0.52720	299.0
Downcomer to Ten Jet Pumps (Suction)	4	1.48200	299.0
Ten Jet Pumps to Lower Plenum	19	11.0800	98.00
Steam Line - See Figure A-1.1			
Steam Dome to Steam Lines	50	5.6730	595.0
One Relief Valve	51	.09501	61.00
Two Relief Valves	52	.19002	61.00
One Relief Valve	53	.09501	61.00
Two Safety Valves	54	.18640	61.00

TABLE A-1.1 (Continued)

VY ATWS RETRAN MODEL INFORMATION

Component Description	Number	Junction Area (ft^2)	Elevation (in)
Steam Line (Continued)			
Steam Line Upstream of MSIVs	55	5.6730	43.00
Main Steam Isolation Valves	56	5.6730	160.6
Steam Line One Downstream of MSIVs	57	5.6730	-29.00
Turbine Stop Valves	58	1.0000	-81.00
Turbine Bypass Valves	59	1.0000	-74.80
Steam Line Two Downstream of MSIVs	60	5.6730	-39.50
Steam Line Three Downstream of MSIVs	61	5.6730	-50.40
Injection System - See Figure A-1.1			
* Steam Flow to HPCI Turbine	(997	1.0000	61.00)
* High Pressure Coolant Injection	(998	1.0000	378.5)
* Feedwater Injection	(999	1.0000	378.5)

TABLE A-1.2

VY ATWS RETRAN Model Safety and Relief Valves

Number of Valves	RETRAN Junction Number	Set Pressure (psia)	Opening Pressure (psia)	Closing Pressure (psia)	Capacity at Set Pressure (lb/hr)
R/V 1	51	1095	1095	1062	800,000
2	52	1105	1105	1072	807,000 each
1	53	1115	1115	1082	815,000
S/V 2	54	1245	1282	1245	925,700 each

* Information bounded by parentheses was incorporated specifically for the VY ATWS RETRAN model.

TABLE A-2.1

RELAP5YA MODEL INFORMATION

Region or System	Nodalization ID Numbers	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDPVOL	Active	TMDPJUN	Active	Passive
Reactor Pressure Vessel							
Lower Plenum	002 to 012	6	0	10	0	0	6
Control Rod Guide Tubes	022 to 024-5	6	0	6	0	0	6
Core Bypass	100 to 102-9	10	0	10	0	0	10
Core Fuel Assemblies							
116 Peripheral Low Power	120 to 122-9	10	0	11	0	9	10
248 Central Average Power	140 to 142-9	10	0	11	0	9	10
4 Central High Power	160 to 162-9	10	0	11	0	9	10
Upper Plenum	206 to 208	2	0	2	0	0	2
Standpipes and Separators	210 to 228	3	3	5	2	0	3
Intermediate Steam	232	1	0	2	0	0	0
Steam Dryer Assembly	234	1	0	1	0	0	1
Steam Dome	240	1	0	1	0	0	1
Downcomer	250 to 290	15	0	14	0	0	13
Other Systems							
Recirculation Loop A	302 to 338-1	12	0	12	0	0	12
Jet Pump Bank A	340 to 342-3	4	0	6	0	0	4
Recirculation Loop B	252 to 338-1	12	0	12	0	0	12
Jet Pump Bank B	390 to 392-3	4	0	6	0	0	4
Feedwater Lines	400 to 402-3	3	1	3	1	0	0
Main Steam Lines	540 to 550	6	1	7	0	0	5
Steam Relief and Supply	551 to 571	0	1	6	1	0	0
Containment	614 to 663	0	8	0	0	0	0
ECCS	700 to 763	2	4	2	4	0	0
DEG	801 to 802	0	0	2	0	0	0
TOTALS		118	18	140	8	27	109

Table A-2.2

RELAP5YA SAFETY/RELIEF AND SAFETY VALVES SETPOINTS

Component Identifier	Junction Id	Opening Pressure (psid)	Closing Pressure (psid)
Safety/Relief Valve 1	551	1,080.0	1,047.6
Safety/Relief Valves 2&3	553	1,090.0	1.057.3
Safety/Relief Valve 4	555	1,100.0	1,067.0
Safety Valves 1&2	557	1,240.0	1,202.8

All Safety/Relief Valve junctions close if ADS open.

Table A-2.3

RELAP5YA CONTAINMENT MODEL INITIAL CONDITIONS

Parameter	Value	
Drywell Pressure	16.6	psia
Drywell Temperature	165	^oF
Drywell Atmosphere	Nitrogen	
Drywell Humidity	100	%
Wetwell Pressure	14.7	psia
Wetwell Temperature	100	^oF
Wetwell Liquid Volume	69,000	ft^3

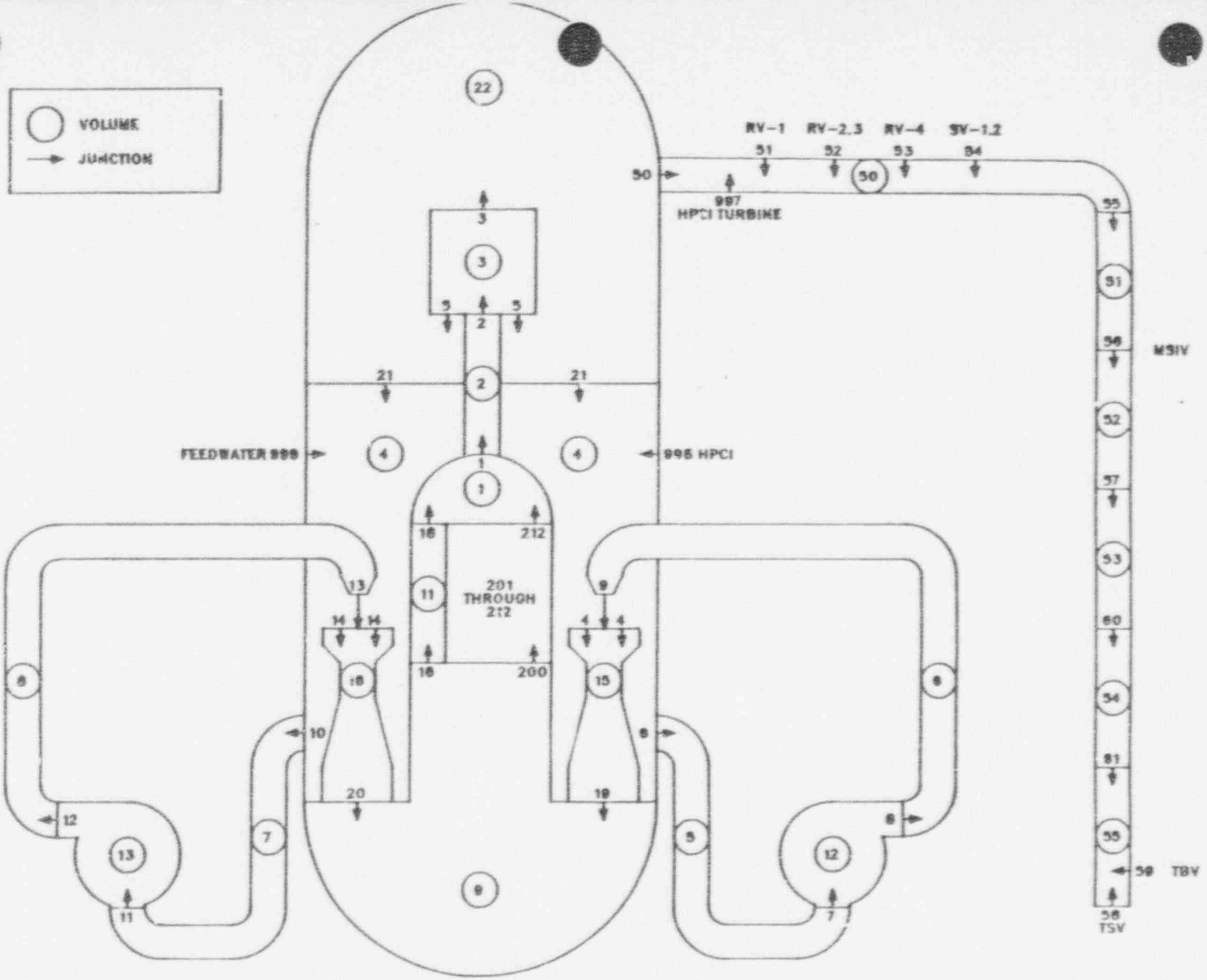


Figure A-1.1 VY ATWS RETRAN Model

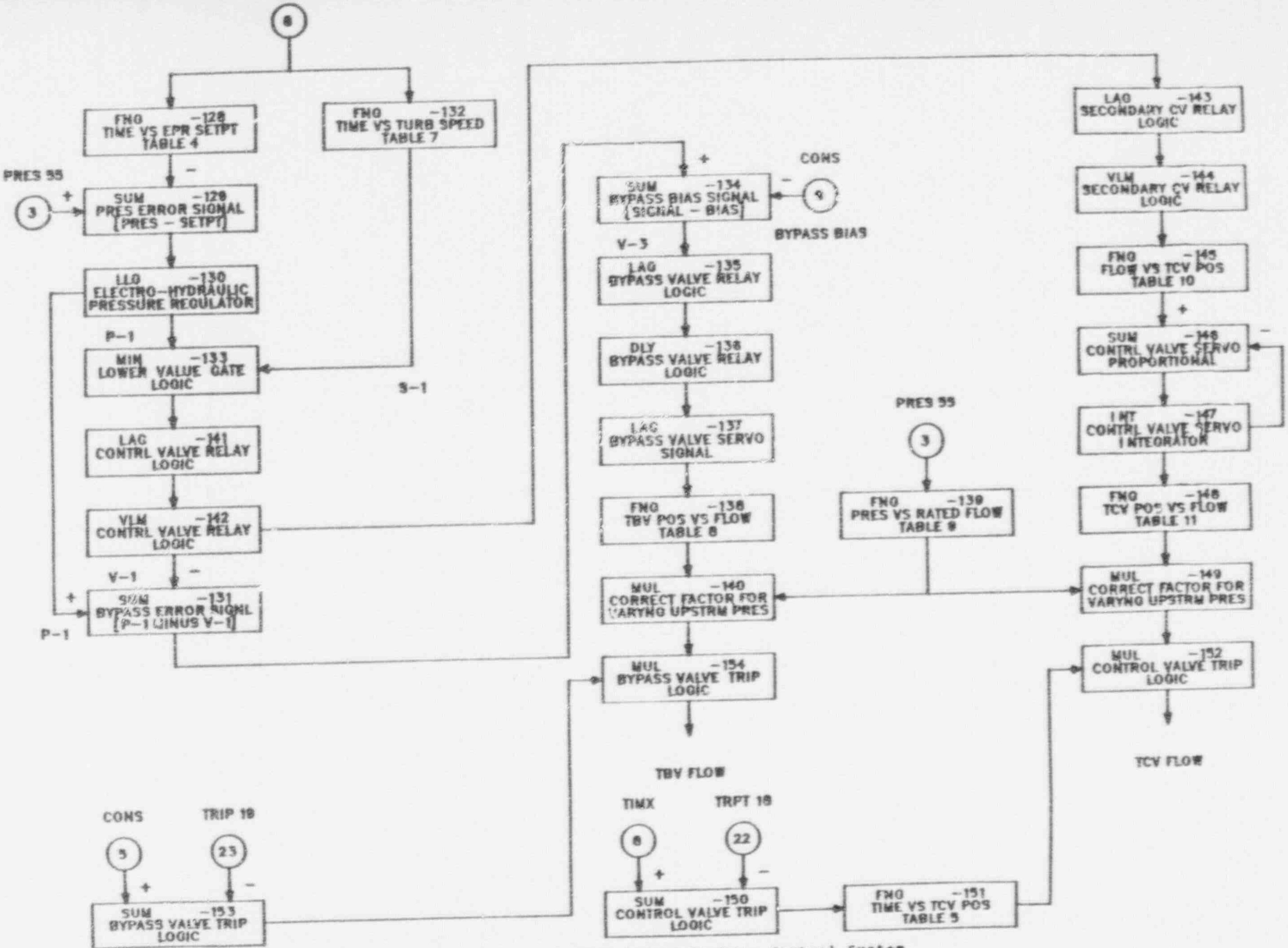


Figure A-1.2 VY ATWS RETRAN Turbine Control System

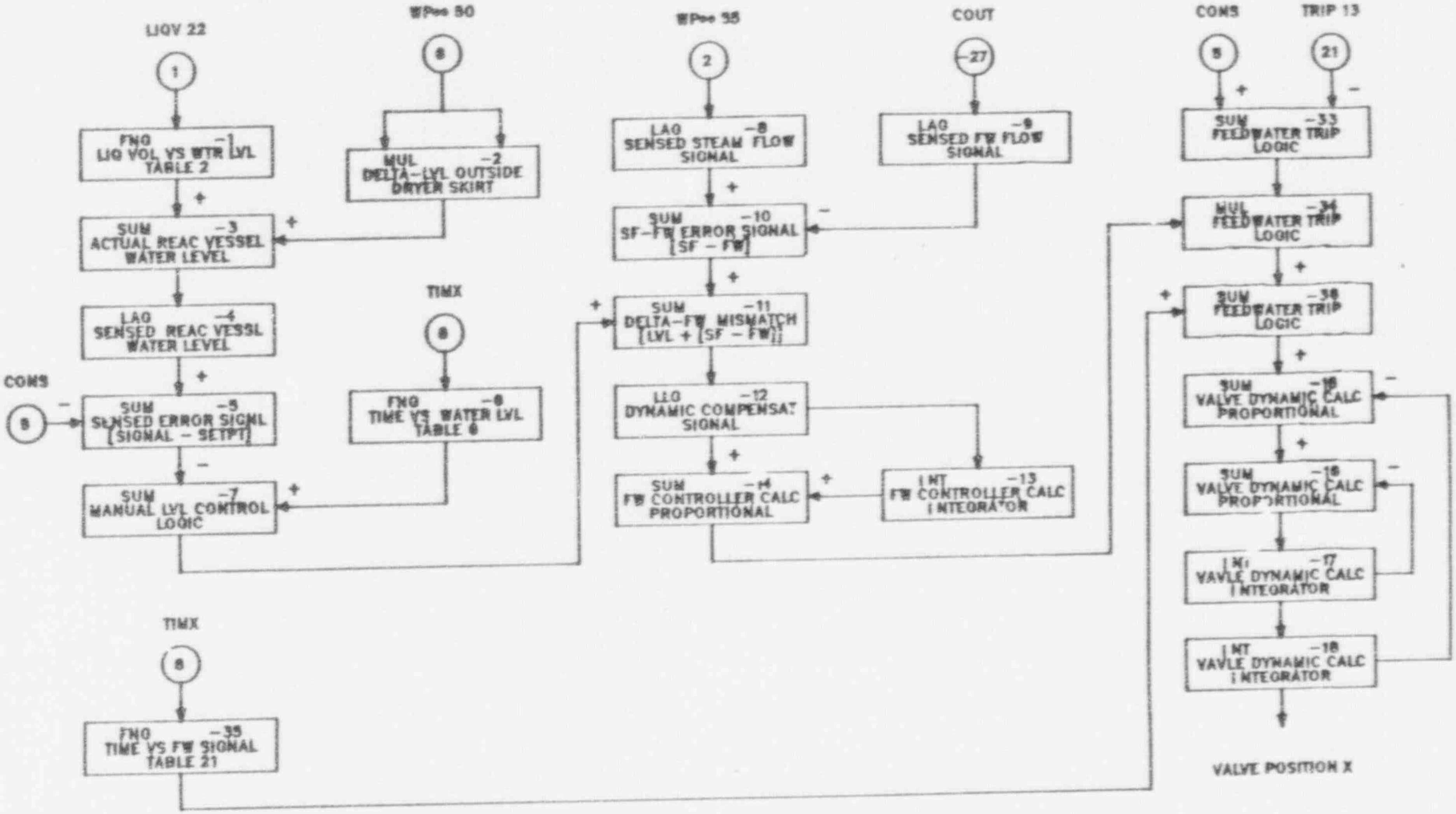


Figure A-1.3 VY ATWS RETRAN Feedwater Control System (Sheet 1 of 2)

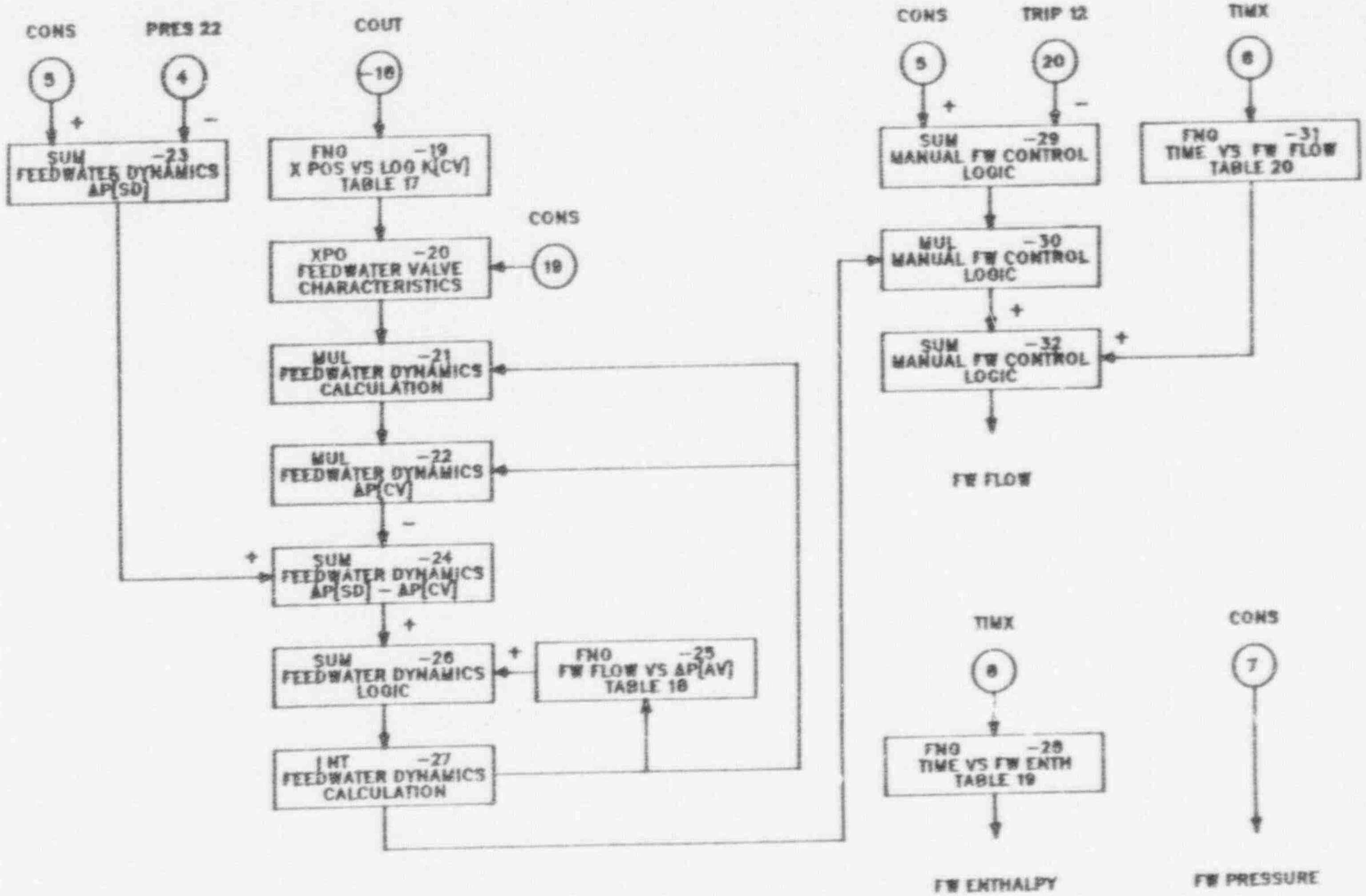


Figure A-1.3 VY ATWS RETRAN Feedwater Control System (Sheet 2 of 2)

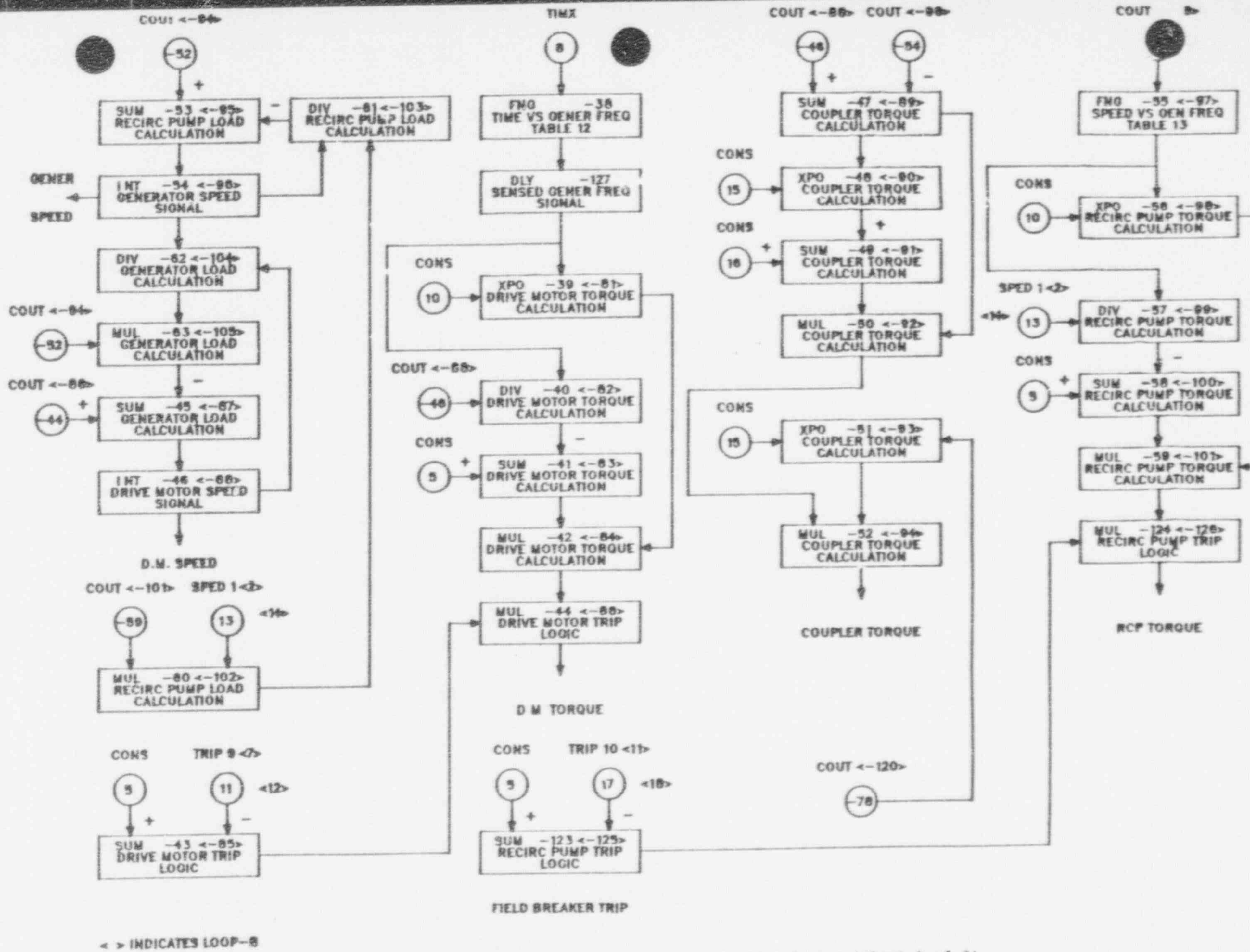


Figure A-1.4 VY ATWS RETRAN Recirculation Control System (Sheet 1 of 2)

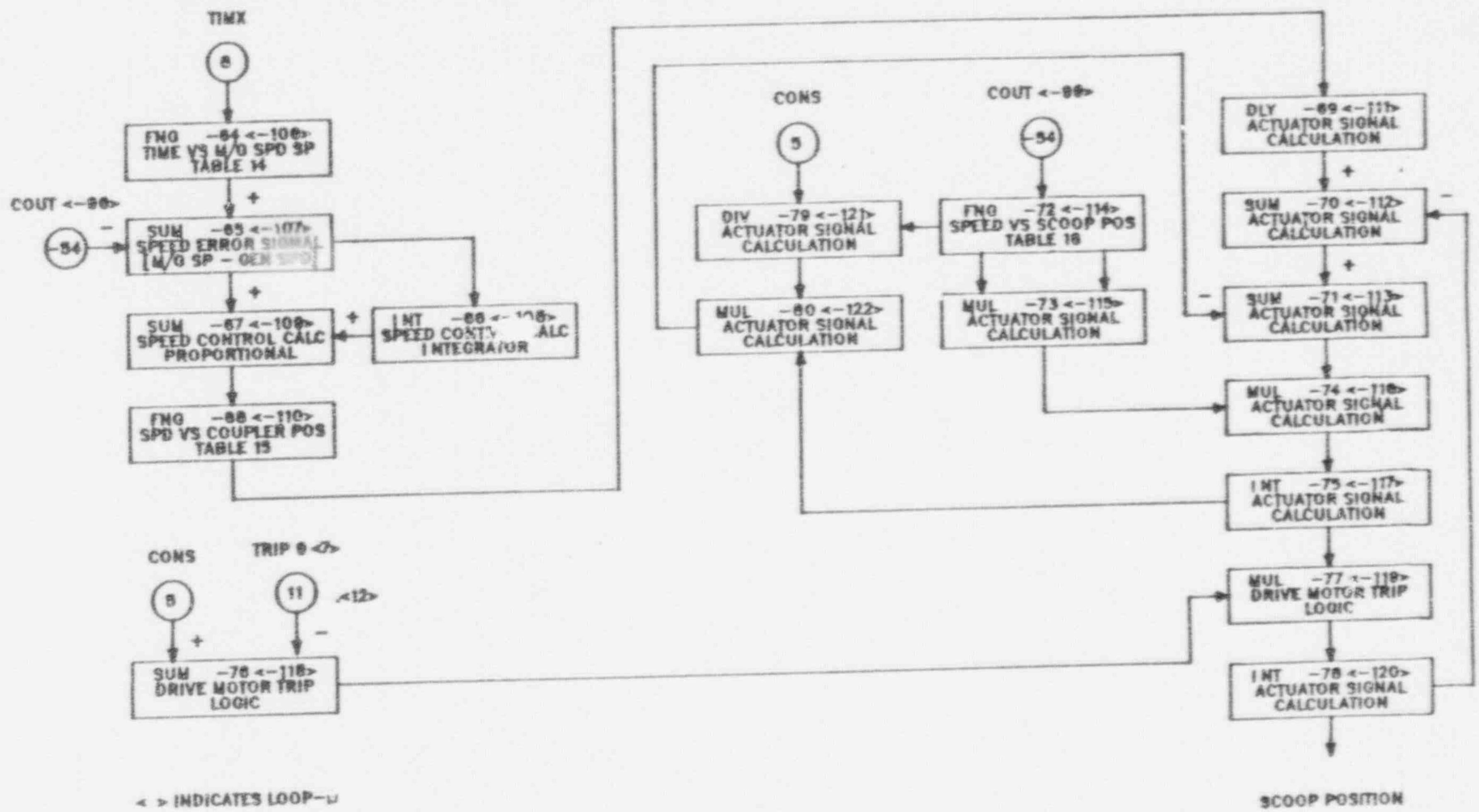


Figure A-1.4 VY ATWS RETRAN Recirculation Control System (Sheet 2 of 2)

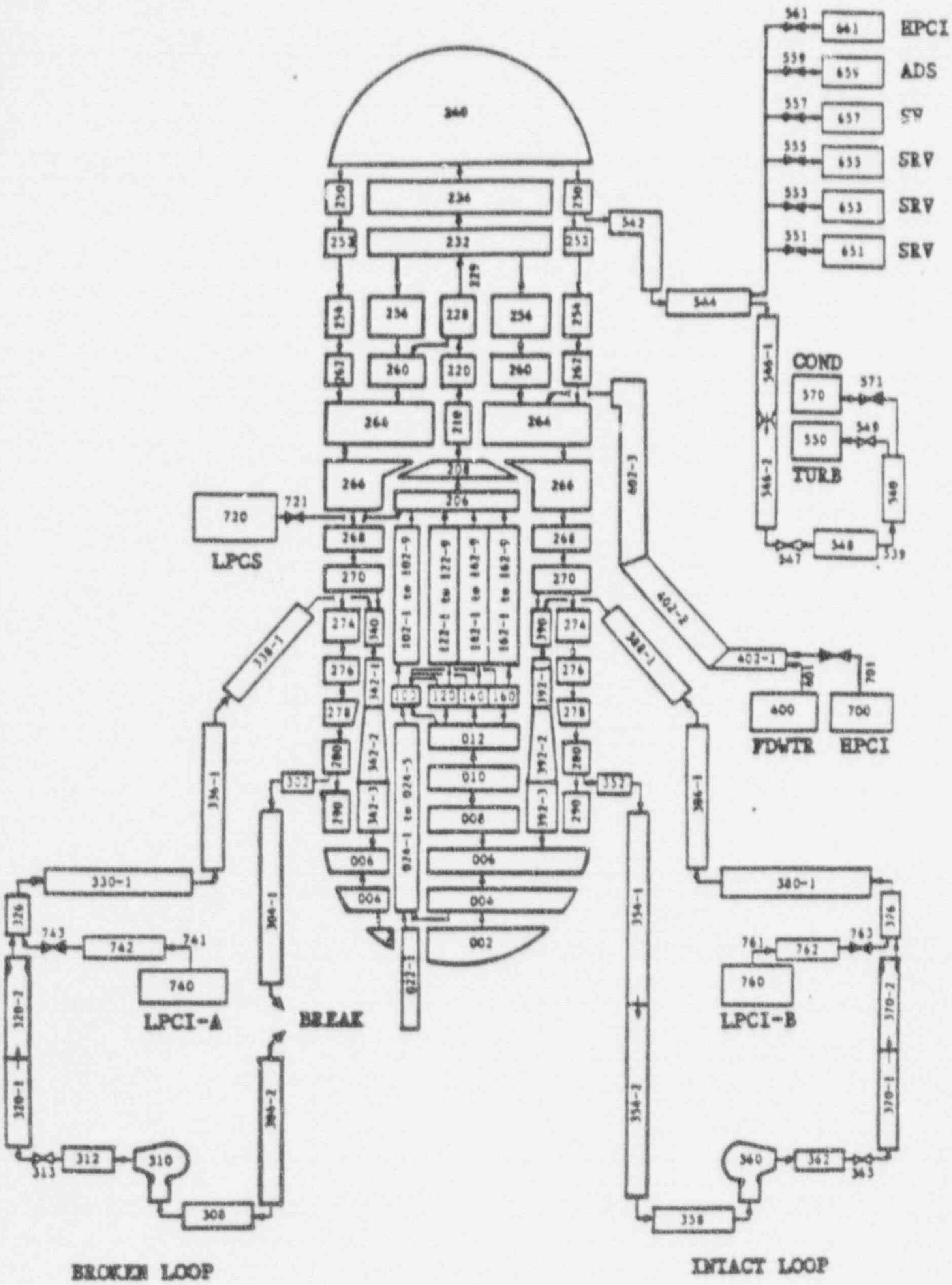


Figure A-2.1 VY LB-LOCA RELAP5YA NSSS Model

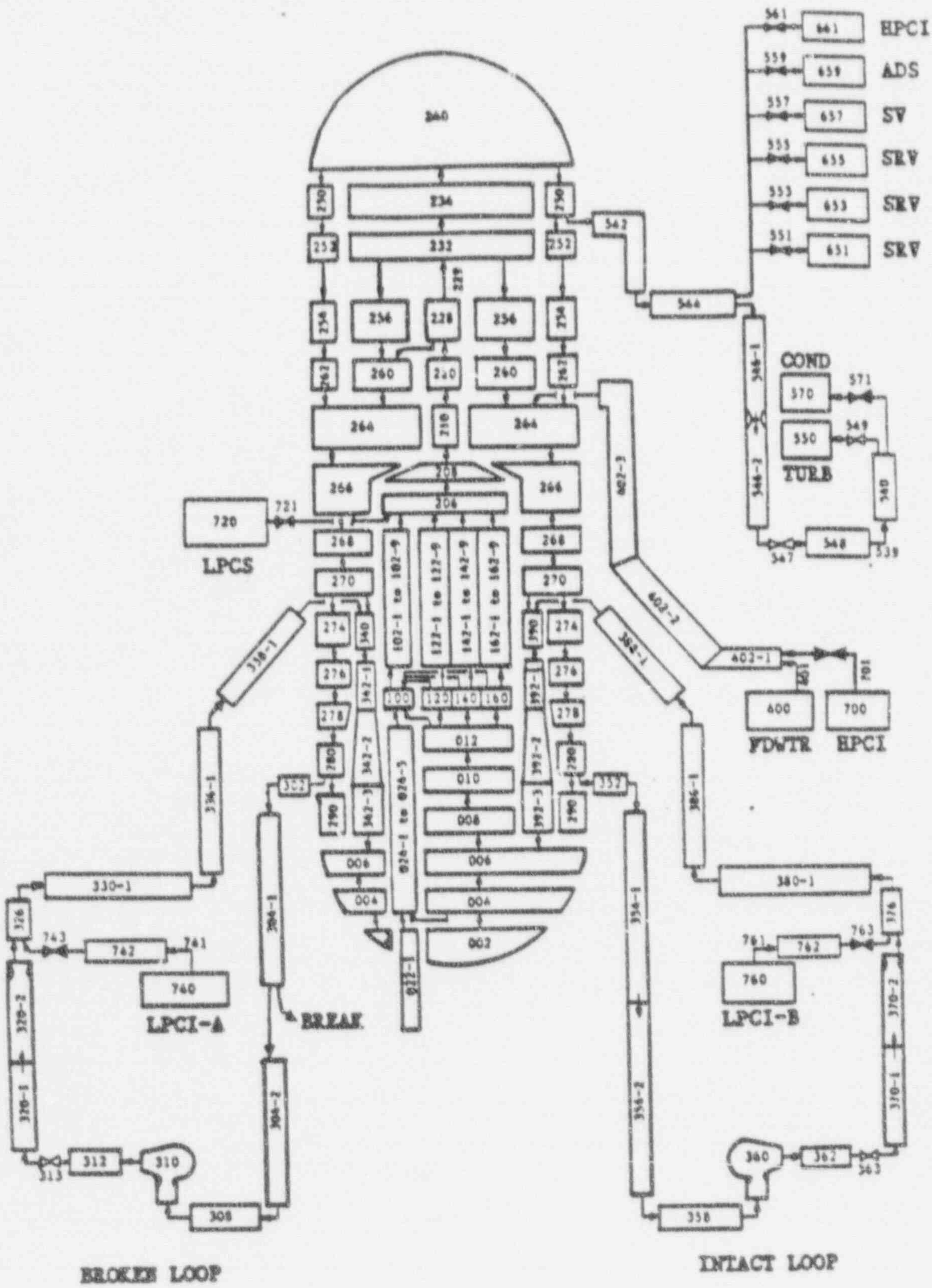


Figure A-2.2 VY SB-LOCA RELAP5YA NSSS Model

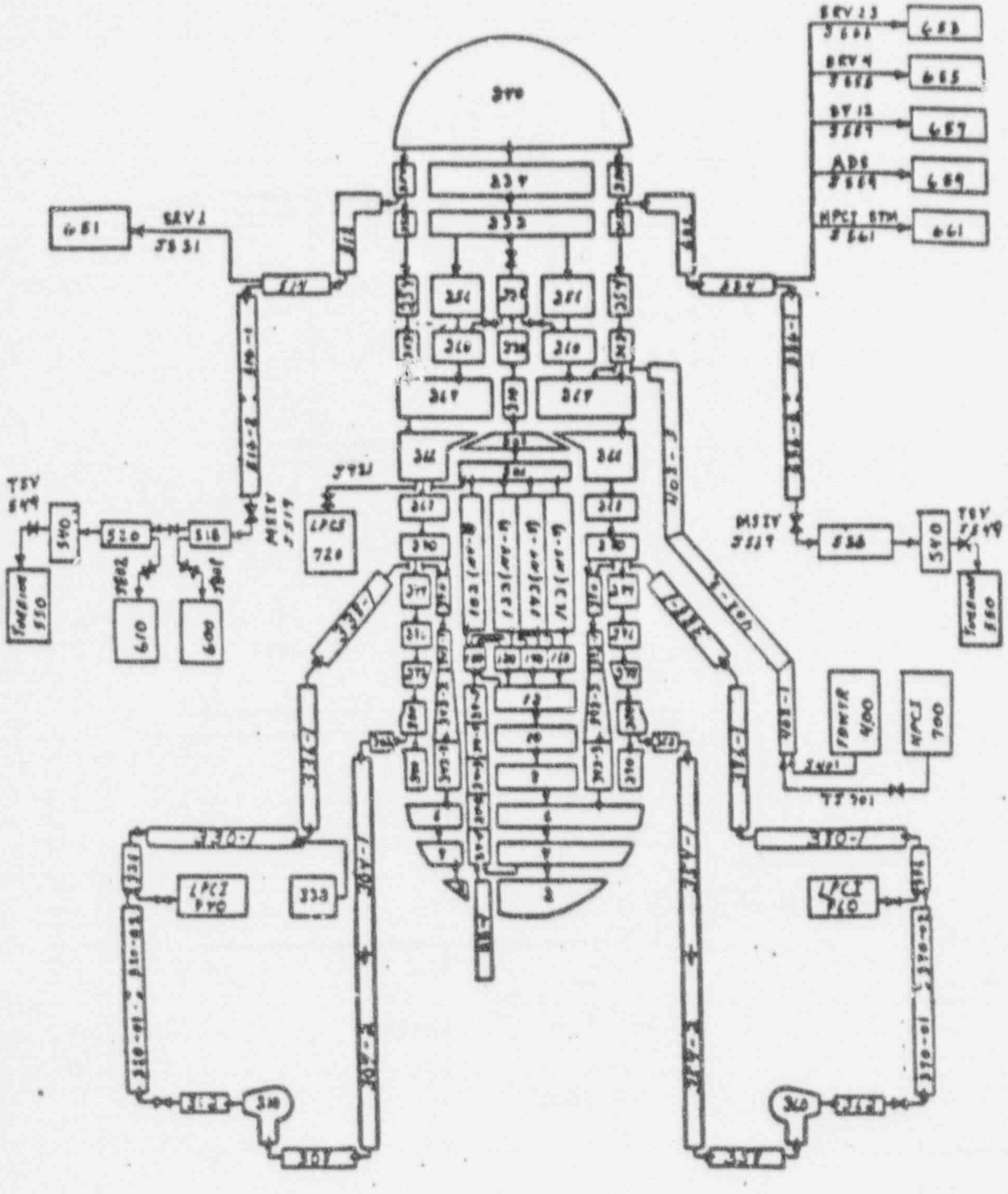


Figure A-2.3 VY MSLB RELAP5YA NSSS Model

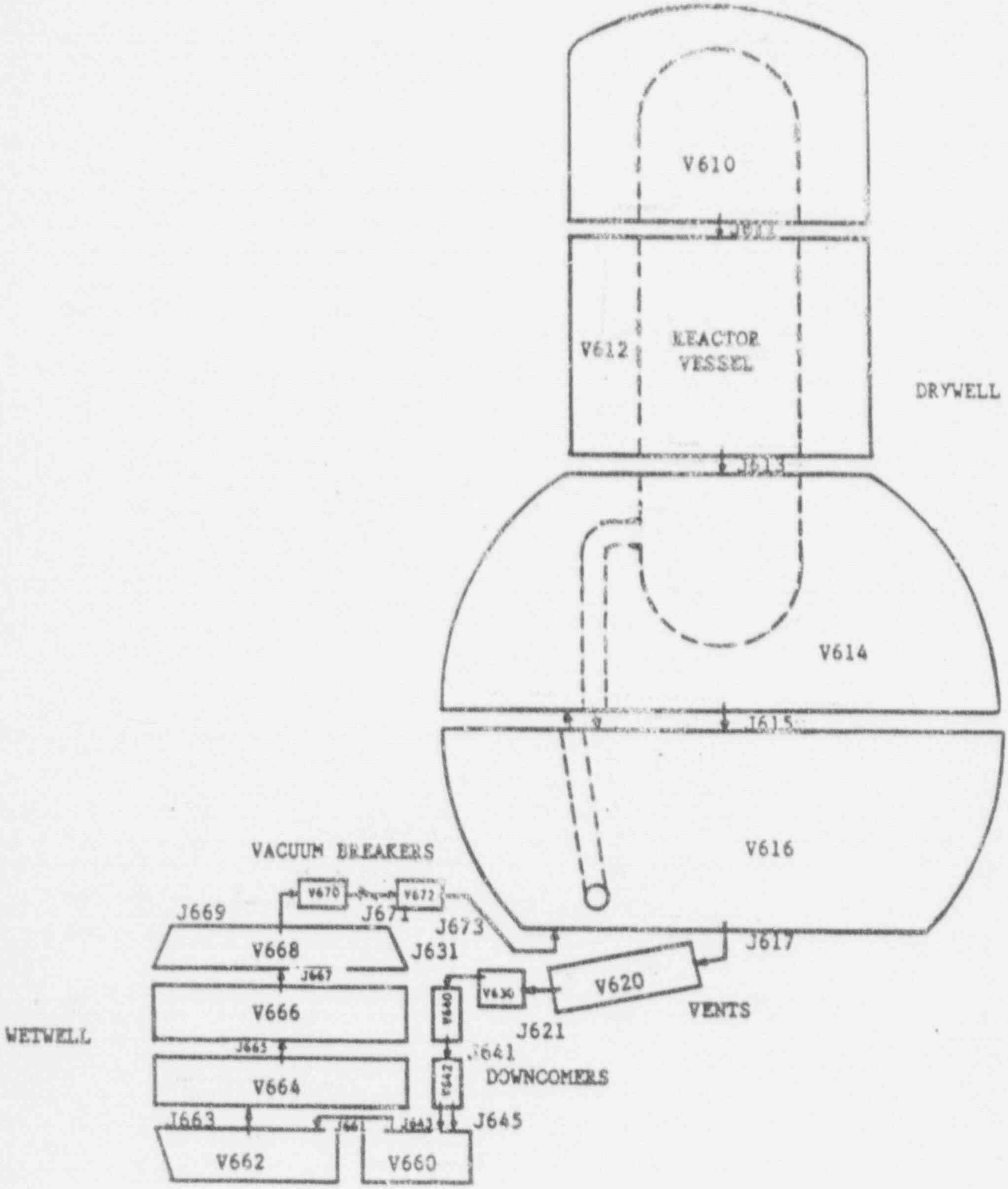


Figure A-2.4 VY RELAP5YA Containment Model

VERMONT YANKEE
MALFUNCTION CAUSE AND EFFECTS TABLE OF CONTENTS

VERMONT YANKEE
MALFUNCTION CAUSE AND EFFECTS TABLE OF CONTENTS

VERMONT YANKEE NUCLFAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Relief Valve Leak (RV2-71A-D)

NUMBER: AD01

TYPE: Generic (A,B,C,D), Variable (0 - 100% = 0 - 50% Valve Position)

CAUSE: Selected RV2-71A-D, Not Properly Seated

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause the relief valve (RV2-71A-D to leak. The magnitude of the leak will be dependent on severity selected. The affected valve's discharge pipe temperature will increase. A temperature greater than 200°F will actuate alarm, "Safety or Blowdown Valve Leakage". Any increase on primary containment pressure, suppression pool temperature, will depend on leakage flow rate and duration malfunction is active. If automatic signals should occur requiring the affected valve(s) to open, the valve will actuate as required.

Removal of the malfunction will restore the affected valve to normal.

REFERENCES: GEK-32437, Automatic Blowdown System
OP 2122
CWD B-191301, sh. 750 - 756

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Relief Valve Stuck Open (RV2-71A-D)

NUMBER: AD02

TYPE: Generic (A,B,C,D)

CAUSE: Selected RV2-71A-D, Mechanically Binds in the Open Position

PLANT CONDITIONS:

Reactor High Pressure Condition

EFFECTS: This malfunction when activated will not prevent relief valves from opening in either auto or manual mode but will prevent the selected valves from closing once opened. The affected relief valve(s) will continue to discharge main steam to the suppression pool. Suppression pool level and temperature will continue to increase. A red open indicating light on CRP 9-3 vertical display will light, indicating the relief valve(s) are open. Simultaneously, the green shut indicating light will go out and an alarm, "RX Relief Valve Open", on CRP 9-3 will annuciate.

If the turbine is operating at high load conditions, the effect will be a loss of generator load as the turbine control valves close to maintain reactor pressure due to the increased steam flow.

If reactor pressure is being controlled via the bypass valves, they will close down or shut completely in an effort to maintain reactor pressure.

If the MSIVs are shut and reactor pressure is sufficiently high enough, a reactor cooldown and blowdown will occur.

Removal of the malfunction will restore the affected valve(s) to normal.

<u>REFERENCES</u>: GEK-32437, Automatic Blowdown System
OP 2122
CWD B-191301, sh. 750 - 756

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Relief Valve Fails to Open (RV2-71A-D)

NUMBER: AD03

TYPE: Generic (A,B,C,D)

CAUSE: Selected RV2-71A-D, Solenoid Failure

PLANT CONDITIONS:

Reactor Operating at Power, 0 - 100%

EFFECTS: This malfunction will cause the selected relief valve(s) not to open on an actuation solenoid operate signal. The red open light on CRP 9-3 will illuminate indicating relief valve(s) are open but the valve(s) will remain closed. If main steam line or vessel pressure exceeds its lift setpoint the selected valve(s) will open.

Removal of the malfunction will allow the relief valve(s) to operate normally.

REFERENCES: CWD B-191301, sh. 752 - 755

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Inadvertent ADS Actuation

NUMBER: AD04

TYPE: Discrete

CAUSE: Spurious Signal Actuating Relays 2E-K6A/B, 2E-K7A/B

PLANT CONDITIONS:

Plant Startup or Power Operation

EFFECTS: This malfunction will result in the automatic depressurization system actuating inadvertently. Actual indication and annunciation for drywell pressure, ECCS pumps, and reactor water level will not be affected. All relief valves will fully open (RV2-71A-D). Both A and B logic push buttons must be depressed to reset the seal-in auto actuation logic once the malfunction is deleted.

This will result in a rapid blowdown of the reactor vessel. Initially reactor water level will swell due to rapid depressurization of the RPV, resulting in trip of the turbine due to high level. Subsequently the Rx will scram on turbine control valve closure signal. Reactor level will immediately decrease resulting in RCIC and HPCI initiation, ECCS initiation. The RHR and Core Spray systems will inject into the Rx vessel. RCIC and HPCI will not inject due to high level trip.

The Rx will completely depressurize due to ADS valves remaining open throughout the transient. Rx water level will be restored by the condensate, RHR and CS pumps.

Removal of the malfunction will restore relays 2E-K6A/B, 2E-K7A/B to normal, allowing the ADS timers to be reset.

REFERENCES: GEK-32437 Automatic Blowdown System
OP 2122
CWD B-191301, sh. 750 - 756

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: ADS Timer Failure

NUMBER: AD05

TYPE: Discrete

CAUSE: Time Delay Contacts (K5A and K5B) Remain (en

PLANT CONDITIONS:

Loss of Coolant Accident

EFFECTS: This alfunction will prevent the ADS valves from opening when the automatic initiation signals are present. The valves will open from manual initiation or mechanically due to high pressure.

If a LOCA condition exists and the malfunction is activated, the rate at which the reactor is depressurized will be significantly longer. The low pressure ECCS systems will be unable to provide water to the Rx vessel until it is depressurized sufficiently.

The rate of depressurization and level drop will be consistent with mass and energy balances on the vessel.

The removal of the malfunction will restore ADS to normal operation.

REFERENCES: GEK-32437 Automatic Blowdown System
CWD B-191301, sh. 750 - 756

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Relief Valve Fails to Open (RV2-71A-D) (Mechanical)

NUMBER: AD06

TYPE: Generic (A,B,C,D)

CAUSE: Selected RV2-71A-D, Mechanically Binds

PLANT CONDITIONS:

Reactor High Pressure Condition

EFFECTS: This malfunction will cause the selected relief valve(s) not to open under any circumstances, automatic signal or manual initiation. If vessel pressure exceeds their lift setpoint, the selected valve(s) will not open. Depending on the number of relief valves selected and reactor pressure, a safety valve actuation may occur.

Removal of this malfunction will allow the relief valve(s) to operate normally.

REFERENCES: CWD B-191301, sh. 752 - 755

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Annunciator System Failure

NUMBER: AN1PW

TYPE: Generic

CAUSE: Circuit Failure Causing Individual Annunciator Window Not to Actuate

PLANT CONDITIONS:

Any Plant Conditions

EFFECTS: The specified annunciator window(s) will be extinguished and will not actuate on any trip signals. Annunciators for the affected window(s) will become silent. No other effects will be seen.

Removal of this malfunction will cause all annunciator windows in an alarm condition to actuate, regardless of their status before this malfunction was active. Alarm conditions which clear while this malfunction is active will not be indicated.

REFERENCES: CWD B-191301, sh. 72 - 83

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Spurious Annunciator Actuation

NUMBER: AN2PW

TYPE: Generic

CAUSE: Circuit Failure Causing Spurious Input to the Annunciator Alarm Panel

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The specified annunciator window will alarm as if its alarm condition was present if power is available to the annunciator panel. The alarm will silence and acknowledge normally.

There is no effect on other systems as the result of this malfunction. For example, an annunciator for a condition which should cause a pump trip will not cause a pump to trip.

Removal of the malfunction will clear the alarm if an actual alarm condition does not exist.

REFERENCES: CWD B-191301, sh. 72 - 83

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Pump (A,B,C) Trip

NUMBER: CD01

TYPE: Generic (A,B,C)

CAUSE: Activation of Instantaneous Overcurrent Device (50)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected condensate pump(s) to trip due to activation of its overcurrent device. The affected condensate pump discharge flow and pressure will decrease. Loss of flow through the affected pump causes the condensate header pressure to decrease until other running or standby condensate pumps can reestablish normal flow.

If an insufficient number of condensate pumps are running to supply the required condensate flow condensate header pressure will decrease causing a feed pump trip on low suction pressure.

All associated annunciators will actuate when appropriate.

If the pump is not running when the malfunction is inserted, it will trip when started.

Removal of the malfunction will allow the condensate pump to be restarted.

REFERENCES: P+ID G-191167
OP 2170
CWD B-191301, Sh. 530-532

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: DELETED

NUMBER: CD02

TYPE:

CAUSE:

PLANT CONDITIONS:

EFFECTS:

REFERENCES:

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Demineralizer (A,B,C,D,E) Resin Depletion

NUMBER: CD03

TYPE: Generic (A,B,C,D,E) Variable (100% = No Conductivity Reduction)

CAUSE: Old Resin

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected condensate demineralizers to decrease its ability to remove ionic impurities. The affected condensate demineralizer effluent conductivity will increase to a value dependent on hotwell conductivity. Condensate demineralizer trouble annunciator will actuate.

Reactor water cleanup influent conductivity and recirc sample conductivity will increase at a rate that is dependent on feedwater mass flow rate. The affected condensate demineralizer may be removed from service via an IDA and a standby demineralizer placed in service.

When a demineralizer has been removed from service, 15 minutes is required for full efficiency of the demineralizer to be reestablished.

REFERENCES: OP 2171 P+ID
G-1911574, G-1911274

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Transfer Pump (A,B) Trip

NUMBER: CD04

TYPE: Generic (A,B)

CAUSE: Activation of Thermal Relay Device (49)

PLANT CONDITIONS:

Any Plant Cordition

EFFECTS: This malfunction will cause the selected condensate transfer pump, if running or when started, to trip due to activation of the thermal overload device. The following equipment will be affected: Condensate system pressurizing line, condenser vacuum pump, RHR, CS system flushing and off gas drain seals. All associated annunciators will actuate when appropriate.

Removal of the malfunction will reset the overload trip device and allow normal operation of the affected condensate transfer pump.

REFERENCES: P+ID G-191176
CWD B-191301 sh. 523-529

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Demineralized Water Transfer Pump (A,B) Trip

NUMBER: CD05

TYPE: Generic (A,B)

CAUSE: Activation of Thermal Relay Device (49)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected demin water transfer pump, if running or when started, to trip due to activation of the thermal overload device. The following equipment will be affected stator winding cooling water makeup, standby liquid control system makeup. All associated annunciators will actuate when appropriate.

Removal of the malfunction will reset the overload trip device and allow normal operation of the affected demin water transfer pump.

REFERENCES: P+ID G-191176
CWD B-191301 sh. 470-471

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Minimum Flow Valve (FCV-4) Failure

NUMBER: CD06

TYPE: Discrete, Variable (0 - 100% Valve Position)

CAUSE: Controller Output Signal Failure

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the condensate minimum flow valve to fail to the severity selected. The min flow valve automatically provides low flow protection for the condensate pumps and to prevent overheating the SJAE's and steam packing exhauster whenever steam is being admitted to them.

If the min flow valve was initially open maintaining minimum flow requirements and the min flow valve is closed, the condensate pump discharge pressure will increase to the shut off head value. Pump amps will decrease to minimum no load value. Condensate recirc low flow annunciator will actuate on CRP 9-6. The loss of condensate flow, as cooling medium for SJAE after and intercondenser and the steam packing exhauster will cause higher than normal operating pressures. All associated annunciators will actuate as appropriate.

Removal of the malfunction will restore the controller output signal to normal and restart the condensate minimum flow valve (FCV-4) to normal operation.

REFERENCES: P+ID G-191157
CWD B-191301 sh. 530

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Storage Tank Leak

NUMBER: CD07

TYPE: Discrete, Variable (100% = 5000 gpm)

CAUSE: Manhole Flange Loose

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the condensate storage tank to leak at the severity selected. As the level decreases, condensate transfer system level trouble annunciator will actuate. Actual level indicator on CRP 9-6 and CRP 9-3 will reflect this fluid loss. If level is allowed to continue to decrease the following equipment will lose suction. Core Spray Pumps, Condensate Transfer Pumps, RCIC Pump, HPCI Pump, CRD Pumps and Condensate Makeup Line. All associated annunciators will actuate when appropriate.

Removal of the malfunction will stop the leak.

REFERENCES: P+ID G-191176
CWD B-191381 sh. 528

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Low Pressure Feedwater Heater (3A,4A,5A,3B,4B,5B) Tube Leak

NUMBER: CD08

TYPE: Generic (3A,4A,5A,3B,4B,5B) Variable (100% - 150% Drain Capacity

CAUSE: Tube Failure

PLANT CONDITIONS:

Reactor Operating at Power, Turbine On-Line

EFFECTS: This malfunction will cause the selected low pressure feedwater heater tubes to leak at the severity selected. This results in a loss of feed/condensate to the affected heater shell. As the shell level increases the normal level control valve will open. If the leak rate is higher than the drain rate, level will increase until the emergency level control valve opens to the condenser. Low pressure feedwater heater high level annunciator will actuate. If the affected heater reaches the Hi-HI level setpoint, the associated extraction non-return valve will close and the extraction steam dump valve will open. The associated heater drain pump will auto start on high level in heater No. 4 if in Auto.

Feedwater heater temperature will decrease resulting in a decrease of feedwater temperature at the heater outlet and will be transmitted through the string resulting in an increase in reactor power but a loss of plant efficiency.

Removal of the malfunction will restore the low pressure feedwater heater(s) tubes to normal.

REFERENCES: P+ID G-191157, 191158
RP 2170

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Low Pressure Heater (A,B) String

NUMBER: CD09

TYPE: Generic (A,B)

CAUSE: Low Pressure Heater Train Inlet Valve Switch (9-6-46, 9-6-47) Close contacts (3-3T) Weld close

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the selected LP HTR train inlet valve to close. The selected valve closed indicating light on CRP 9-6 will illuminate. Reactor feedwater inlet temperature will decrease causing an increase in core inlet subcooling. Due to the negative void reactivity coefficient, an increase in core power results. Reactor water level will be maintained by feedwater reg. valves. Reactor feedwater pump suction pressure will decrease due to isolation of the heater string. Condensate system flow, pressure and pump amps will reflect the loss of this flow path. All associated annunciators will actuate when appropriate.

Opening of the LP heater bypass valve will return the condensate system pressure and flow to normal.

Removal of the malfunction will restore the closing contacts to normal and allow the affected LP heater string to be returned to normal operation.

REFERENCES: P+ID G-191157
CWD B-191301, Sh. 533-534
RP 2170

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Exhaust Hood Spray Valve Fails Closed

NUMBER: CD10

TYPE: Discrete

CAUSE: Loss of Air Signal To Spray Valve

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will result in the exhaust hood spray valve going to its fully closed position. Spray pressure will decrease to zero. Hood temperature will begin to increase. At any time exhaust hood temperature exceeds 175°F its appropriate annunciator will actuate on CRP 9-7. Provision is made to manually regulate this temperature by a motor operated bypass valve controlled on CRP 9-23. If exhaust hood temperature reaches 225°F, turbine trip will occur.

Removal of malfunction will restore the turbine exhaust hood spray valve to normal operation.

REFERENCES:

P+ID G-191157
GE Turbine Technical Manual
CWD B-191301, Sh. 130, 151

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Hotwell Level Controller Failure

NUMBER: CD11

TYPE: Generic (A,B) Variable (0-100% of Controller Output)

CAUSE: Failure of Controller Auto Output

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause the selected hotwell level controller auto output to fail to the severity selected.

If the malfunction sets the hotwell level controller auto output signal to minimum the normal reject valve will close and the makeup valve will open, allowing water to be transferred from the CST. Actual hotwell level will begin to increase. The emergency reject valve will open at 84%. If level in the hotwell continues to increase, and the condenser tubes start becoming covered condenser vacuum will start decreasing. The turbine will eventually trip on low condenser vacuum.

If the malfunction sets the hotwell level controller auto output signal to maximum the normal makeup will close, if open, and the normal reject valve will open, allowing water to be rejected to the CST. Actual hotwell level will begin to decrease. Condensate storage tank level will start increasing. If the level is allowed to continue to decrease, it will result in condensate pump trip, reactor feed pump trip and reactor scram due to low reactor water level.

At any time during this malfunction, the operator may place the affected level controller in manual or select the unaffected controller and reverse the transient if plant conditions permit. All associated alarms will actuate when appropriate.

Removal of this malfunction will restore the affected controller auto output signal to normal.

REFERENCES:

P+ID C-191157
CWD B-191301, Sh. 518

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condensate Filter Demineralizer (A,B,C,D,E) Resin Break-Through

NUMBER: CD12

TYPE: Generic (A,B,C,D,E)

CAUSE: Selected Condensate Filter (A-E) Retention Element and Resin Trap Failure

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the selected condensate filter demin's (A,B,C,D,E) retention elements/resin trap to fail resulting in resin breakthrough. The following major events will be evident over a short time:

1) Condensate system effluent conductivity will remain at approximately normal values.

2) Delta-P across the associated demin's resin trap increases to 10 PSI, causing condensate demin trouble alarm on CRP 9-6 to actuate.

3) Slight increase in RFP suction pressure indication/slight decrease in condensate pump pressure.

4) Increased RWCU system influent conductivity.

5) Off-gas flow increases.

Removal of malfunction will restore affected filter demineralizer to normal.

REFERENCES:

NEDO-25016
P+ID G-191157, 191274

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Hotwell Level Transmitter

NUMBER: CD13

TYPE: Discrete, Variable (0-100% = Range of Transmitter)

CAUSE: Failure of Level Transmitter LT-1A

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: If the "A" condenser Hotwell is selected, at 100% this malfunction will cause condenser hotwell reject valves LCV-1A-2/LCV-1A-3 to fully open due to failure of level TX LT-1A to maximum output. Upon malfunction activation, the condenser hotwell reject valves(s) will open fully, if closed. As hotwell level decreases, appropriate annunciators will actuate as events occur. The hotwell makeup valves will auto-open prior to receiving low level alarm. If the makeup valves are in manual and closed, then hotwell will continue to decrease and CST level will increase. If the hotwell level is allowed to continue to decrease, then NPSH requirements for the condensate pump(s) cannot be met and a subsequent trip of the condensate pump(s) will result. The RFP will trip on low suction pressure and reactor water level will quickly decrease to scram setpoint.

If the "A" condenser Hotwell is selected, at 0% this malfunction will cause condenser hotwell reject valves LCV-1A-2/LCV-1A-3 to close due to failure of level TX LT-1A to minimum output. An excessively high hotwell level will result in main condenser vacuum degradation.

Removal of the malfunction will return TX LT-1A to normal and allow the condenser hotwell reject valves to operate normally.

REFERENCES:

P+ID G-191157, Sh. 1
CWD B-191301, Sh. 75

VERMONT YANKEE NUCLEAR POWER CORPORATI .

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Core Spray Pump (A,B) Trip

NUMBER: CS01

TYPE: Generic (A,B)

CAUSE: Actuation of Instantaneous Overcurrent Trip Device (50)

PLANT CONDITIONS:

Any Plant Conditions

EFFECTS: This malfunction will cause the selected core spray pump, if running, to trip. If the pump is supplying water to the reactor vessel, vessel level will reflect the loss of flow from the respective core spray loop. The affected loop pressure and flow will decrease. All associated annunciators will actuate when appropriate.

Removal of the malfunction will allow restart of the affected core spray pump.

REFERENCES: P+ID G-191168
CWD B-191301, sh. 1164

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Core Spray Pump (A,B) Fails to Auto Start

NUMBER: CS02

TYPE: Generic (A,B)

CAUSE: Auto Start Relay 14A-K12A or K12B, Contacts (1-2) fail to close

PLANT CONDITIONS:

Conditions Requiring Core Spray Pump Auto Start

EFFECTS: This malfunction will result in the core spray pump not starting when an auto initiation signal is present. The operator may start the pump manually and provide coolant to the reactor vessel.

Removal of the malfunction will allow the pump to start automatically if an auto initiation signal is present. If the pump is already running, no effects will be seen from either activation or removal of this malfunction.

REFERENCES: GEK-9612
CWD B-191301, sh. 1150 - 1151, 1163 - 1164

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Core Spray Injection Valve (12A, 12B) Fails to Auto Open

NUMBER: CS03

TYPE: Generic (A,B)

CAUSE: Auto Start Relay 14A-K12A or K12B, Contacts (1-2) Fail to Close

PLANT CONDITIONS:

Conditions Requiring Core Spray System Initiation

EFFECTS: This malfunction will allow the core spray system equipment to operate normally on an initiation signal with the exception of valve 12A and/or 12B remaining closed. This will prevent water from being admitted to the reactor. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. The operator can manually open the affected valve from CRP 9-3 and restore flow to the vessel.

Removal of the malfunction will restore core spray injection valve(s) to respond to auto initiation signal.

REFERENCES: P+ID G-191168
GEK-9612
CWD B-191301, sh. 1167

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Core Spray (A,B) Pipe Break Inside RPV, Between RPV and Shroud

NUMBER: CS04

TYPE: Generic (A,B)

CAUSE: Pipe Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected core spray pipe to break inside RPV between RPV wall and shroud. This will allow differential ressure switches to sense jet pump differential p essure instead of core differential pressure. When the differential pressure exceeds approximately 4 psid the high differential pressure annunciator will actuate, no other effects will be seen unless the core spray system is injecting water into the vessel.

If core spray is in progress, the affected loop flow will be diverted to the shroud's downcomer annulus instead of the core spray header. Reactor water level will reflect the change of this water supply and will depend on remaining equipment in operation.

Removal of the malfunction will require reinitialization of the Simulator.

<u>REFERENCES:</u> GEK-9612
P+ID G-191168

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU Pump (A,B) Trip

NUMBER: CU01

TYPE: Generic (A,B)

CAUSE: Activation of Thermal Relay Device (49)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: If the pump is running, or when it is started, the thermal relay device (49) will actuate. The indicating lights will continuously show the breaker status. The breaker will trip, the pump will stop, and the pump tripped annunciator will actuate.

System flow will decrease to zero and pump discharge pressure will go down to the value of suction pressure. With system low flow condition present the cleanup holding pumps will start. Temperatures in the system will gradually decay to ambient temperature and the heat load on RBCCW will decrease rapidly. Conductivity indications will increase slowly due to stagnant water. Actual vessel conductivity will increase slowly due to the buildup of corrosion products.

Removal of the malfunction will restore normal operation of the pump. The system will have to be operated for a period of time proportional to the time the system was stopped before equilibrium conductivity will be re-established in the reactor vessel.

REFERENCES: CWD B-191301, sh. 903 - 904
P+ID G-191178 sh. 1 - 2
OP 2122

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU Non-Regenerative Heat Exchanger Tube Leak

NUMBER: CU02

TYPE: Discrete, Variable (100% = 100 gpm at normal differential pressure)

CAUSE: Tube Erosion at Inlet Tubesheet

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: Water will flow from the RWCU system into the RBCCW system. Level in the RBCCW surge tank will increase actuating the high level annunciator on CRP 9-6 and then overflow to the reactor building floor drain system. The leak will cause increased heat load and activity in the RBCCW system which will in turn result in increased RBCCW and RWCU temperatures. The leak will also cause more influent flow through the regenerative heat exchanger which will result in higher inlet temperatures to the NRHX. Process radiation monitors will detect any increase in activity.

When temperature out of the NRHX exceeds 140°F the RWCU system will isolate. The pumps will trip as the isolation valves close. RWCU system pressure will remain relatively constant until the system is isolated. RWCU system will then depressurize to RBCCW system pressure.

Removal of the malfunction will stop the leak.

REFERENCES: OP 2122
P+ID G-191178, sh. 1

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU Leak Upstream of Regenerative Heat Exchanger

NUMBER: CU03

TYPE: Discrete, Variable (100% = 500 gpm at Normal Differential Pressure)

CAUSE: Pipe Break at Influent Inlet to Regenerative Heat Exchanger

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The leaking water will flash to steam as it is reduced to atmospheric pressure the temperature in the RWCU HX room will increase rapidly. As the temperature exceeds 115°F, "Steam leak Panel Temperature High" annunciator on CRP 9-4 will actuate. When area temperature exceeds 125°F, the system will isolate. The room temperature may be observed on CRP 9-21. Process radiation and area radiation monitoring will detect any increase in activity.

At 100% severity, if system flow rate at discharge of the filter demineralizers drop below minimum flow a pump trip will occur. See effects of malf CU01. When the RWCU system is isolated, the syst . will depressurize rapidly.

After RWCU system is isolated and malfunction removed, the leak will stop.

REFERENCES: OP 2122
P+ID G-191178, sh. 1

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU Drain Flow Controller Failure

NUMBER: CU04

TYPE: Discrete, Variable (0 - 100% Valve Position)

CAUSE: PCV-55 E/P Converter Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: RWCU drain flow valve CU-55 will go to the instructor specified position.

If used to bleed away excess water during reactor startup and hot standby operation and the valve is selected to close, Rx water level will begin to increase. The rate of increase will be a function of the heatup rate and flow rate from CRD system. If the valve is selected to open, RWCU system pressure will decrease and drain flow will increase. Rx water level will begin to decrease.

Cleanup discharge Hi-Lo pressure annunciator actuates at 5 psig decreasing or 140 psig increasing. Regenerative heat exchanger temperatures will reflect the change in flow path.

Removal of the malfunction will restore normal operation of the drain flow control valve.

REFERENCES: P+ID G-191178, sh.1
OP 2122

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU Inadvertent System Isolation

NUMBER: CU05

TYPE: Discrete

CAUSE: RWCU Isolation Valve Control Relays (K27+K28) Fail

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause relays (K27+K28) to fail resulting in a RWCU system isolation. RWCU system isolation valves (V12-15, V12-18 and V12-68) close. RWCU system pump breaker will trip due to V12-18 not full open and pump trip annunciator will actuate. The remaining effects due to this malfunction are presented in malfunction effects CU01.

Removal of this malfunction will restore relays K27 and K28 to normal operation which will allow RWCU system to be placed back into operation.

REFERENCES:
CWD B-191301, sh 910 - 912
P+ID G-191173, sh.1
OP 2112

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Line Break in RWCU Pump Room B

NUMBER: CU06

TYPE: Discrete, Variable (100% = 400 GPM)

CAUSE: Pipe Break at "A" and "B" Line Intersect

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause water leaking from the RWCU line to flash to steam as it reaches atmospheric pressure in the RWCU pump area. The leak detection system will indicate and alarm as temperature in the area rises. As the temperature rises above the isolation setpoint, isolation valves CU-15, CU-18 and CU-68 will close. Reactor Bldg area ARMs will respond to the release of radioactive steam.

Removal of the malfunction will cause the leak to stop resulting in decreasing area temperatures and radiation levels.

REFERENCES: OP 2121

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator (A,B) Trip

NUMBER: DG01

TYPE: Generic (A,B)

CAUSE: Lockout Relay 86DG Trip

PLANT CONDITIONS:

Diesel Generator Running

EFFECTS: This malfunction will cause the selected standby diesel generator lockout relay 86DG to trip. The affected diesel generator lockout trip annunciator on CRP 9-8 will actuate. The selected diesel generator output breaker will trip. The diesel engine will stop. If the diesel generator was carrying loads at the time of malfunction, individual systems with equipment on this bus will respond to the power loss. Indications for the affected diesel generator frequency, amperes, voltage, kilowatts will reflect the diesel generator trip.

If the diesel is not running when the malfunction is activated, the lockout will still occur and disable the machine.

Removal of the malfunction will allow the lockout relay to be reset via an IDA and return the affected diesel generator to normal.

REFERENCES: CWD B-191301, sh. 610 - 611
P+ID G-191270

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator (A,B) Slow Start

NUMBER: DG02

TYPE: Generic (A,B) Variable (0% = Normal Start Time) (100% = 11 Additional Seconds)

CAUSE: Governor Control Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected DG governor control circuit to fail to the instructor specified value. When the DG is started, and 100% severity is selected, the diesel will take 11 extra seconds to reach the 810 RPM setpoint when the diesel running permissive is set. This will result in a diesel generator starting failure alarm. If the diesel is already running, at rated speed and/or load, this malfunction will have no effect on the machine operation.

Removal of the malfunction will restore the governor to normal, eliminating the long starting time and return the affected DG to normal operation.

REFERENCES: Fairbank Morse NY 706173
P+ID G-191270

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator A Auto Voltage Regulator Failure

NUMBER: DG03

TYPE: Generic (A,B), Variable A = High (0 to 1 KV), B = Low (0 to -1 KV)

CAUSE: Auto Regulator Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the "A" DG auto voltage regulator to fail to the instructor specified value. Manual operation of the regulator will not be affected by the auto regulator failure. The affected diesel generator auto voltage regulator range indicating failure. If transfer from auto voltage regulator to manual voltage regulator is not made prior to reaching the trip setpoint the affected diesel generator will trip. Refer to DG01 for similar effects. All associated annunciators will actuate when appropriate.

Note: This malf is either/or but not both - active in "auto" model only.

Removal of the malfunction will restore the affected diesel generator auto voltage regulator to normal operation.

REFERENCES: OP 2126
CWD B-191301, sh. 610, 611, 616, 617

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel (A,B) Generator Auto Voltage Regulator Oscillation

NUMBER: DG04

TYPE: Generic (A,B), Variable (100% = plus and minus 5% swing)

CAUSE: Auto Regulator Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected DG auto voltage regulator to fail to the instructor specified value. At 100% severity the generator terminal voltage will vary plus and minus 5% from the automatic regulator setpoint. However, the underexcited or overexcited limits will not be exceeded during the oscillation. The oscillation frequency will be equal to the maximum rate of change of the auto voltage regulator. The auto voltage regulator range indicating lights on CRP 9-8 will flash off and on. The affected diesel generator volt, amp, watt indications on CRP 9-8 will reflect the oscillation. Manual operation of the regulator will not be affected by the auto regulator failure.

Removal of the malfunction will return the auto voltage regulator to normal operation.

REFERENCES: OP 2126
CWD B-191301, sh. 610, 611, 616, 617

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator Fails (A,B) To Start

NUMBER: DG05

TYPE: Generic (A,B)

CAUSE: Air Start Solenoid Valves (AS1 and AS2) Fails to Open

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected diesel generator air start solenoid valves not to open. This results in the selected diesel generator failing to start, both manually or automatically. Eleven seconds following either manual or automatic start signal, the affected diesel generator failure to start, diesel generator shutdown and diesel generator aux trouble annunciators on CRP 9-8 actuate.

If the diesel is already running, no effects will be seen from this malfunction.

Removal of the malfunction will restore the affected DG air start solenoid valves to normal operation.

REFERENCES: Fairbanks Morse NY 706173
P+ID G-191270

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator (A,B) Speed Control Failure

NUMBER: DG06

TYPE: Generic (A,B), Variable (100% = plus and minus 1% change)

CAUSE: Governor setpoint drift (Oscillation)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected diesel generator governor setpoint to drift to the instructor specified value. This malfunction affects the governor only after the engine is operating and has no effect during starting of the engine.

If the diesel is not paralleled, the speed of the machine will oscillate plus or minus 9 rpm (.6 hertz). Otherwise, the oscillation will be plus or minus 10% of rated load or 300 kw. If the change is such that all load is shed from the machine, the breaker 9-8 will reflect DG trip. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore the governor to normal operation, and reset any relays tripped as a result of this malfunction.

REFERENCES: Fairbanks Morse NY 706173
CWD B-191301, sh. 610 - 613
OP 2126

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Diesel Generator (A,B) Low Lube Oil Pressure

NUMBER: DG07

TYPE: Generic (A,B), Variable (0% = Normal Pressure, 100% Fully Clogged Strainer)

CAUSE: Lube Oil Strainer Plugged

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected DG lube oil strainer to become plugged to the instructor specified value. If 100% severity is selected the affected diesel generator lube oil pressure low annunciator on CRP 9-8 will actuate at 20 psig. The diesel generator will trip at 16 psig. The effects listed in DG01 are similar. All associated annunciators will actuate when appropriate.

If the diesel is not running when the malfunction is activated, no effects will be seen.

Removal of the malfunction will restore the affected DG plugged strainer to normal operation.

REFERENCES: Fairbanks Morse NY 706173
OP 2126

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Unit Auxiliary Transformer No. 2

NUMBER: ED01

TYPE: Discrete

CAUSE: Failure of Transformer No. 2 Differential Relay 87/UT

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in the loss of unit auxiliary transformer No. 2 due to actuation of primary lockout relay 86/G-P caused by failure of transformer No. 2 differential relay 87/UT.

Appropriate Transformer lockout annunciators will actuate. When 86/G-P Trips, the following actions are initiated:

a. Opening of 345 KV generator breakers 81-1T and 1T

b. Opening of Generator exciter field breaker

c. Tripping of turbine by main trip solenoid

d. Tripping of 4 KV unit auxiliary breakers 12 and 22

e. Signaling the loss of normal power (LNP) detection circuiting, effecting fast transfer of in-house loads.

f. Energizing two auxiliary relays (86X/GP and 86X-1/GP - both type GE-HEA relays) which in turn initiate:

1) Opening of generator 345 KV disconnect switch (T-1 MOD)
2) Stopping of auxiliary transformer #2 cooling fans and pumps
3) Stopping of generator stator cooling pumps
4) Signaling the computer - D632

Removal of the malfunction will allow restoration of unit auxiliary transformer No. 2 to normal conditions once 86/G-P is manually reset.

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Startup Transformer

NUMBER: ED02

TYPE: Generic: A,B

CAUSE: Failure of Transformer 3A or 3B Sudden Pressure Relay 63 FP-A/B

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of selected startup transformer 3A or 3B due to actuation of the primary lockout relay trip (GE-HEA, 86/ST) caused by the failure of transformer 3A or 3B sudden pressure relay 63 FP A/B. Appropriate transformer lockout annunciators will actuate. When 86/ST trips, it initiates the following actions:

A. Trip 4160 volt feeder breakers #13, #23 and #53 and block reclosing of these breakers unless the relay is reset.

B. Open 115 KV auxiliary lockout relays 86/STX-A,B to trip 115 KV oil circuit breaker K1.

C. Blocks the resetting of 87 STA/X, 87 STB/X relays (87 STA/X and 87 STB/X are used to stop transformer coolers.)

D. Reclosing of 115 KV oil circuit breakers K1, is allowed after disconnect switch T-3 MOD opens.

Removal of the malfunction will allow restoration of startup transformer 3A or 3B to normal condition.

REFERENCES: P+ID G-191298, sh. 1,2; G-191299
VYNPC Simulator Malfunction Cause and Effects

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Normal 4160V Bus (1,2,5A,5B)

NUMBER: ED03

TYPE: Generic Bus (1,2,5A,5B)

CAUSE: Bus Fault Due to Ground

PLANT CONDITIONS:

Reactor Startup or Operating at Power

EFFECTS: This malfunction will result in the loss of the selected normal 4160V bus (Bus No. 1,2,5A,B) due to bus fault. Upon malfunction activation the supply circuit breakers to the affected 4160V bus will trip, associated 4.16KV voltage indication will go to zero volts appropriate annunciators for bus trip will actuate. Interlocks prevent alternate feed to the affected bus as long as the fault has the bus tripped. The following major loads will be de-energized.

A. 4160V Bus No. 1

1. Reactor Feedwater Pump 1A
2. Reactor Feedwater Pump 1B
3. Condensate Pump 1A
4. Circulating Pump 1A
5. Reactor Recirc MG Set 1A
6. Station Service Transformer T11-1A
7. Station Service Transformer T-6-1A

B. 4160V Bus No.2

1. Reactor Feedwater Pump 1C
2. Condensate Pump 1B
3. Condensate Pump 1C
4. Circulating Pump 1B
5. Circulating Pump 1C
6. Reactor Recirc MG Set 1B
7. Station Service Transformer T-10-1A
8. Station Service Transformer T-7-1A
9. Breaker No. 22

C. 4160V Bus No. 5A

1. Breaker No. 53
2. Subsequent Loss of 4160V Bus No. 5B

D. 4160V Bus No. 5B

1. Circ Water Booster Pump 1A
2. Circ Water Booster Pump 1B
3. Circ Water Booster Pump 1C
4. Station Service Transformer T-581-1A
5. Station Service Transformer T-582-1A

Removal of the malfunction will reset the 4160V bus fault, then re-energization of the selected bus may be performed and lost loads returned to normal.

<u>REFERENCES</u>: P+ID G-191299, 4KV Auxiliary One-Line Wire Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Emergency 4160V Bus 3,4

NUMBER: ED04

TYPE: Generic (Bus 3,4)

CAUSE: Bus Fault, Due to Ground

PLANT CONDITIONS:

Reactor Startup or Operating at Power

EFFECTS: This malfunction will result in the loss of the selected emergency 4160V bus (Bus No. 3,4) due to bus fault. Upon malfunction activation the supply circuit breakers to the affected 4160V bus will trip. Associated 4.16KV voltage indication will go to zero volts. Appropriate annunciators for bus trip will actuate. Interlocks prevent alternate feed to the affected bus as long as the fault has the bus tripped. The following major loads will be de-energized.

A. 4160V Bus No. 3

1. RHR Service Water Pump 1B
2. RHR Service Water Pump 1D
3. Station Service Water Pump 1B
4. Station Service Water Pump 1D
5. Core Spray Pump 1B
6. RHR Pump 1C
7. RHR Pump 1D

8. Station Service Transformer T-8-1A
9. Breakers 3T1, 3V

B. 4160V Bus No. 4

1. RHR Service Water Pump 1A
2. RHR Service Water Pump 1C
3. Station Service Water Pump 1A
4. Station Service Water Pump 1C
5. Core Spray Pump 1A
6. RHR Pump 1A
7. RHR Pump 1B
8. Station Service Transformer T-9-1A
9. Breakers 4T2, 4V

Associated diesel generators will start automatically on loss of normal power. However, the respective D/G breaker will not close.

Removal of the malfunction will reset the 4160V bus fault. The re-energization of the selected bus may be performed and lost loads returned to normal.

REFERENCES: P+ID G-191299, 4KV Auxiliary One-Line Wire Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of 480v Bus (No. 6,7,8,9,10,11)

NUMBER: ED05

TYPE: Generic (Bus No. 6,7,8,9,10,11)

CAUSE: Bus Fault, Due to Ground

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of selected 480v bus due to bus ground fault. Upon malfunction activation the selected 480v bus (Bus No. 6,7,8,9,10,11) supply ACB will trip open and the alternate feed breaker(s) will be unable to close in due to fault on bus. Generally, current indications will go to zero. Loads on that bus will be de-energized and all appropriate annunciators for bus trip will actuate.

In lieu of a load list, appropriate schematic and/or one line diagrams will be utilized to determine loads from each bus.

Removal of malfunction will reset the fault on the bus and allow its loads to be re-energized.

REFERENCES: G-191303, 480v Auxiliary One Line Wiring Diagram Sht 1
G-191304, 480v Auxiliary One Line Wiring Diagram Sht2
VY-E-40-001-4, Auxiliary One Line Wiring Diagram Sht1
104-00811, 480v Spec Reg./Gen Reg.
104-00794, ED Lesson Plan

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

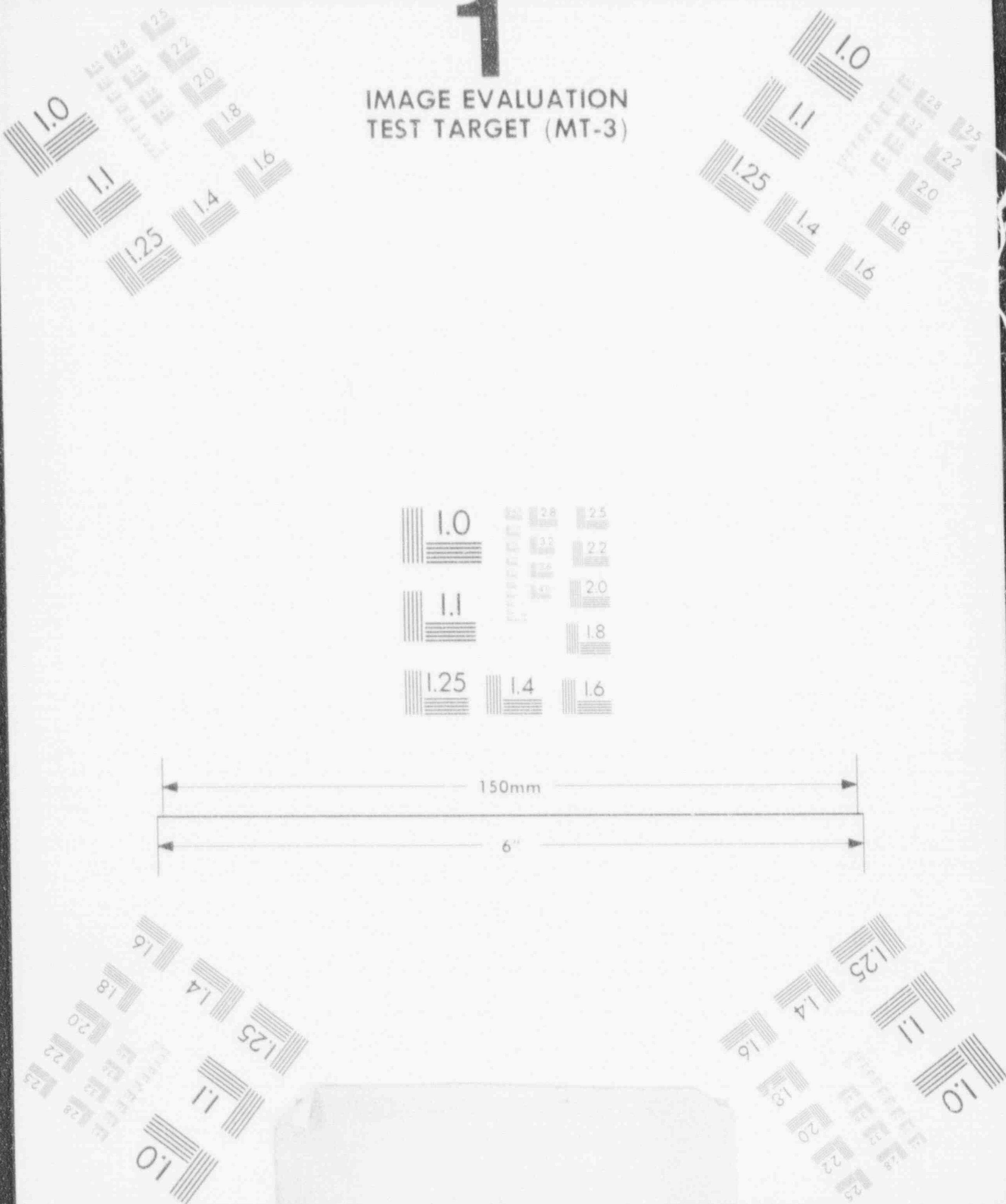
IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)



VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of 125V DC Bus (DC-1,2,3)

NUMBER: ED06

TYPE: Generic (DC-1, DC-2, DC-3)

CAUSE: Bus Fault, Due to Ground

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the loss of the selected 125V DC bus due to bus fault. When the 125V DC buses are de-energized, appropriate annunciators and/or equipment will trip. There are three 125V DC distribution buses. DC-1, DC-2, DC-3. All control and position indication will be lost for those breakers and valves deriving control power from the selected bus. Specific loads affected are referenced in the 125V DC and Vital AC One Line Wiring Diagram.

Removal of the malfunction will return to the 125V DC bus selected to normal.

REFERENCES: P+ID G-191372, 125V DC and Vital AC One Line Wiring Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of 24V DC Bus (A,B)

NUMBER: ED07

TYPE: Generic (A,B)

CAUSE: Bus Fault, Due to Ground

PLANT CONDITIONS:

Any Plaut Condition

EFFECTS: This malfunction will result in the loss of the selected 24V DC bus (A,B) due to bus ground fault. Appropriate annunciators will actuate for bus trouble. There are two distribution panels (buses): Panel A, Panel B. All loads deriving power from the selected 24V DC bus will be de-energized. Specific loads lost are listed on the 24V DC one line diagram.

Removal of the malfunction will restore the selected 24V DC bus to normal.

REFERENCES: P+ID G-191372, 24V DC One Line Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of ECCS 24V DC Bus (A,B)

NUMBER: ED08

TYPE: Generic (A,B)

CAUSE: Bus Fault, Due to Ground

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of the selected ECCS 24V DC bus (A,B) due to ground fault. ECCS 24V DC buses provide power to recirc pump trip analog trip system (RPT), which will cause the affected side pump to trip.

Removal of the malfunction will restore ECCS 24V DC bus (A,B) to normal, allowing restoration of affected reactor recirc pump to service.

REFERENCES: P+ID G-191297, 24V DC One Line Diagram
CWD 850, 851, 855 & 856

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: UPS (A,B) Failure

NUMBER: ED09

TYPE: Generic (A,B)

CAUSE: Loss of Logic Power Supply to Inverter (Blown Fuse)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: . This malfunction will result in the loss of UPS (Uninterruptible Power System) due to failure of logic power supply to the selected inverter (caused by blown fuse). UPS is a 480V AC system that feeds MCC-89A (89B). The associated UPS inverter(s) converts the DC power to AC power to supply their critical loads. Upon malfunction activation the critical load will continue to be powered by the inverter from the battery until the battery is discharged to the point of UPS trip (approximately 15 minutes). If the malfunction is deleted before low DC voltage UPS trip, the charger will restart automatically and the critical load will continue to be supplied from the inverter.

Removal of malfunction will restore the selected UPS to normal.

REFERENCES: VY-E-40-001-4 480V Auxiliary One Line Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Instrument AC

NUMBER: ED10

TYPE: Discrete

CAUSE: Overcurrent Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of the 120/240 VAC instrumentation distribution panel due to overcurrent trip. All loads supplied by the instrument AC bus will be de-energized. When the malfunction is active, there will be no auto-transfer to the emergency supply. Refer to P+ID G-191372 for specific loads affected by this scenario.

Removal of the malfunction will return instrument AC bus to normal with all loads re-energized.

REFERENCES: P+ID G-191372 125 VDC and Vital AC One-Line Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Vital MG Set

NUMBER: ED11

TYPE: Discrete

CAUSE: Breaker Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of vital MG (Vital 120/240 VAC motor-generator set) due to an output breaker failure. Control power will be lost to the electric pressure regulator, rod select relays, feed water flow control stations, process computer, HPCI suction valve logic resulting in transfer to the torus. A manual transfer switch on CRP 9-8 provides an alternate source of power to the vital source of power to the vital AC bus which can be used to restore power to the loads lost.

Removal of the malfunction will allow restoration of the vital MG to normal and corrective action of returning the vital AC bus to the normal line-up.

REFERENCES: P+ID G-191372, 125 VDC and Vital AC One Line Diagram

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: 4 KV Bus (1,2) Failure to Fast Transfer

NUMBER: ED12

TYPE: Generic (A,B)

CAUSE: Relay Failure

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in the failure of the station service loads to fast transfer from the auxiliary transformer to the startup transformer. When the fast transfer circuit fails to transfer, a residual voltage decay transfer relay will attempt to initiate a transfer to the startup transformer.

This scheme closes the startup breaker when voltage on its related non-safety bus decays to 1000 volts and the auxiliary transformer breaker open. Specific equipment that drops out during a residual voltage transfer and is not picked up when voltage is restored are identified in the appropriate P+ID. If this second transfer fails a loss of normal power will occur and the safety buses will be supplied by the diesel generators. During a loss of normal power (LNP) ACB's 3T1 and 4T2 are tripped by under voltage relays and buses 3 and 4 are energized automatically by their respective diesel generator.

Removal of malfunction will restore fast transfer relay to normal and allowing restoration of station service loads to be fast transferred as required.

REFERENCES: OP 2142, 4KV Electrical System
P+ID G-191299, 4KV Auxiliary One Line Wiring Diagrams

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Grid Disturbance

NUMBER: ED13

TYPE: Discrete, Variable: A = High (60 - 60.5 Hz)
B = Low (58.5 - 60 Hz)

CAUSE: Grid Abnormality

PLANT CONDITIONS:

Reactor Operating At Power, Main Generator On-Line

EFFECTS: This malfunction will result in grid disturbance due to grid abnormality, grid disturbance may vary from 60.5 hertz to as low as 58.5 hertz over a 30 sec period.

If grid disturbance is high, grid voltage will increase proportionally as grid frequency increases. Generator load will decrease as grid frequency and generator frequency is increased. As grid voltage is increased, generator exciter voltage and current will decrease, and amplidyne voltmeter will go toward full boost (Generator MVARS will increase in the negative direction). The generator loading will not be outside of the capability curve. The high grid voltage will affect all buses down to and including the 480V buses.

If grid disturbance is low, grid voltage will decrease proportionally but not greater than 5% over the same 30 second period. Generator load will increase as grid frequency and generator frequency is reduced. As grid voltage is reduced, generator exciter voltage and current will increase, and the amplidyne voltmeter will go toward full back (Generator MVARS will increase in the positive direction). When the system frequency decreases to 59.8 H, the turbine speed indicator will be at 1794 rpm. The decrease in voltage will affect all buses down to and including 480 VAC buses.

The generator will trip after 15 seconds at 58.5 hz.

Removal of malfunction will remove grid disturbance and restore network load to normal conditions over a 30-second period.

REFERENCES: 104-00830, Main Generator System Description

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Main Transformer No. 1 Cooling (Group 1, Group 2)

NUMBER: ED14

TYPE: Generic (A,B)

CAUSE: Thermal Overload 49X-1(2) Contacts Opening

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in the loss of selected main transformer No. 1 cooling group (1,2) due to loss of selected cooler group (1,2) power supply caused by thermal overload 49X-1(2) contacts opening. The main transformer No. 1 temperature will increase. The rate of increase will be dependent upon the load on the transformer, the number of cooler groups out of service and the season selected. This will result in a loss of transformer efficiency due to the increase of current losses. The loss of efficiency will be reflected back to generator as indicated by increasing generator voltage, current and MVARs. Actual generator load will remain relatively constant.

Rated load can be maintained for approximately one hour without injury to the transformer with all cooling equipment out of service, following operation at rated load.

Rated voltage at no load can be maintained for approximately six hours without injury to the transformer will all cooling equipment out of service following normal operation at rated load. Approximately seventy percent of rated load can be maintained with a minimum of 50% of the cooling equipment in operation without exceeding the rated temperature rise (55°C - 75°C).

Removal of malfunction restores power to the selected main transformer cooler group (1,2) and subsequent starting of oil pump and associa ed fans.

<u>REFERENCES</u>: GEK-14215
EBASCO Specification 30-62

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Failure of Auto Transformer No. 4

NUMBER: ED15

TYPE: Discrete

CAUSE: V y Inverse Time Overcurrent - Ground Fault

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the failure of Auto Transformer No. 4 due to primary protection relay trip (GE-TAC, 51G/AT) caused by inverse time overcurrent - ground fault. Appropriate transformer annunciators will actuate. When relay 51G/AT trips, the following actions will initiate:

A. Opening of 345 KV North bus breaker T-4.

B. Opening of 115 KV bus breaker K-1.

C. Preventing the 345 KV North bus breaker T-4 from closing and reclosing.

D. Preventing the 115 KV bus breaker K-1 from closing and reclosing.

E. Auto transformer No. 4 megawatts, megavars, and A-C amperes on control room instrumentation reflect no load conditions.

Removal of the malfunction will allow restoration of Auto Transformer No. 4 to normal.

REFERENCES: P+ID G-191298, sh. 1 + 2, G-191299

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Main Transformer No. 1

NUMBER: ED16

TYPE: Discrete

CAUSE: Failure of Main Transformer No. 1 Differential Relay 87/GMT

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in the loss of main transformer #1 due to actuation of the backup lockout relay GE-HEA, 86/G-B caused by failure of main transformer #1 differential relay 87/GMT. Appropriate transformer lockout annunciator(s) will actuate. When relay 86/G-B trips, the following actions are initiated:

A. Opening of 345 KV generator breakers 81-1T and 1T.

B. Opening of generator exciter field breaker.

C. Tripping of turbine by main trip solenoid.

D. Tripping of 4KV unit auxiliary breakers 12 and 22.

E. Signaling the loss of normal power (LNP) detection circuitry effecting a fast transfer.

F. Energizing two auxiliary relays (86X/GB and 86X-1/GB) which in turn initiate:

1. Opening of generator 345KV disconnect switch (T-1MOD).
2. Stopping of main transformer #1 cooling fans and pumps.
3. Stopping of generator stator cooling pumps.
4. Signaling the computer D588.
5. Signaling the 345KV generator breakers (81-1T, 1T).
6. Preventing the 345 KV generator breakers (81-1T, 1T) closing and reclosing.

Removal of malfunction will allow restoration of main transformer #1 to normal once 86/G-B is manually reset.

<u>REFERENCES</u>: 104-16008, Electrical Systems Description
P+ID G-191298, 191299

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Off-Site Power

NUMBER: ED17

TYPE: Discrete

CAUSE: Multiple Faults on Switchyard Due to Severe Storm

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in a total loss of off-site power, resulting in a reactor scram, turbine trip, and generator lockout. Power to all service transformers will be lost. All AC power will de-energize except those which are alternately supplied by DC battery systems. As busses de-energize the emergency diesel generators will auto-start and close onto their respective 4160 VAC busses. Emergency equipment supplied by these busses will sequence back on or will be capable of being started by the operator. The equipment required to bring the reactor to a safe shutdown conditions and maintain it by removing decay heat from the reactor core, will be capable of performing its intended functions.

Removal of the malfunction will restore the switchyard off-site power sources to normal, permitting necessary switching to recover from the event.

REFERENCES: P+ID G-192128-1.2; 191299
FSAR
CWD B-191301, sh. 200-370

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Generator Trip

NUMBER: EG01

TYPE: Discrete

CAUSE: Lockout Relay Trip, 86/G-P

PLANT CONDITIONS:

Reactor Operating at Power with Main Turbine/Generator Synchronized to Grid

EFFECTS: This malfunction will cause the main generator output circuit breaker and field breaker to trip. Indication of phase current, megawatts, megavars will immediately decrease to zero. Appropriate annunciators for a generator trip will actuate. The turbine will trip, with no subsequent reactor scram if the turbine stop valve closure signal to RPS is bypassed due to low power operation. Reactor vessel pressure will be maintained by the bypass valves and relief valves as necessary.

Removal of the malfunction will enable reset of the main generator lockout relay, 86/G-P.

REFERENCES: 104-16008, Electrical Systems Description

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Generator Voltage Regulator Failure

NUMBER: EG02

TYPE: Discrete, Variable; A = High (0 to +2 KV)
B = Low (0 to -2 KV)

CAUSE: Electrical Failure

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will cause the main generator voltage regulator to fail due to an electrical problem in the auto voltage regulator.

If failed low, the following effects will be evident:

A. Main Generator AC Kilovolts Decreases
B. Main Generator Field Amperes Decreases
C. Main Generator DC Field Voltage Decreases
D. Main Generator Amplidyne DC Voltage Goes Toward the Negative Direction
E. Main Generator Kiloamps Increases
F. Main Generator Frequency Remains Constant
G. Main Generator Megawatts Remains Constant
H. Main Generator Megavars Increases Slowly in the Negative Direction
I. 345 KV Bus Voltage Decreases
J. 4 KV Bus Voltage Decreases

If failed high, the following effects will be evident:

A. Main Generator AC Kilovolts Increases
B. Main Generator Field Amperes Increases
C. Main Generator DC Field Voltage Increases
D. Main Generator Amplidyne DC Voltage Increases in the Positive Direction
E. Main Generator Kiloamps Decreases
F. Main Generator Frequency, and Megawatts, Remains Constant
G. Main Generator Megavars Increases Slowly in the Positive Direction
H. 345 KV Bus Voltage Decreases
I. 4 KV Bus Voltage Decreases
J. Voltage Regulator Auto Transfers to Manual Mode

Manual Voltage Adjust may be implemented to stabilize main generator voltage and VAR loading.

Removal of the malfunction will return auto voltage regulator to normal, ready to be transferred from manual to auto.

<u>REFERENCES</u>: Main Generator GEK

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Generator Voltage Regulator Oscillation

NUMBER: EG03

TYPE: Discrete, Variable (0 to 100% = 0 to 50 volts)

CAUSE: Regulator Failure

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will cause the main generator voltage regulator to oscillate due to failure of the auto voltage regulator. Oscillation will be limited to no more than 50 volts at 100% severity. All of the following main generator parameters will reflect oscillation indication:

A. Generator AC Kiloamps
B. Generator AC Kilovolts
C. Generator Field Volts
D. Generator Field Amperes
E. Generator Regulator Transfer DC Volts
F. Generator Megawatts and Frequency will Remain Constant
G. Generator Megavars
H. 345 KV Bus Voltage
I. 4KV Bus Voltage

Manual voltage control may be implemented to stabilize generator voltage and VAR loading.

Removal of malfunction will return the main generator voltage regulator to normal.

REFERENCES: Main Generator GEK

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Generator Load Reject

NUMBER: EG04

TYPE: Discrete

CAUSE: Generator Output Breakers Open Spuriously, Simultaneously

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in a main generator load rejection due to generator output breakers opening spuriously, resulting in significant loss of electrical load on the generator. Major sequence of events are as follows:

A. Generator output breakers open spuriously

B. Generator load drops

C. Turbine control valve fast closure occurs simultaneously as load drops

D. If > 30% power, the reactor scrams on TCV fast closure (if turbine generator load > 45%). If < 30% no scram will result due to TCV and TSV Scram bypass.

Removal of the malfunction will allow turbine generator resync when conditions permit.

REFERENCES: FSAR, Section 14

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Stator Cooling Water Pump (A,B) Trip

NUMBER: EG05

TYPE: Generic (A,B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will cause the selected stator cooling water pump to trip, if it is running. Stator cooling system pressure and flow will decrease, resulting in the automatic startup of the standby pump when pressure decreases 10 psi with a backup at 20 psi.

If the stator cooling system pressure, flow and temperature cannot be maintained, a generator runback will be initiated by the turbine runback circuit in approximately 1 1/2 minutes. The runback signal will be applied by running the speed/load changer back to the specified load. If the load runback should be unsuccessful within 3 minutes after a loss of stator cooling, a signal from the generator protection system will trip the emergency trip system by energizing the master trip solenoid which in turn will open the main generator breakers by sequential tripping.

Removal of the malfunction will allow restart of the affected stator cooling water pump.

REFERENCES: 104-00826, Stator Cooling Water System Description

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: DELETED

NUMBER: EG06

TYPE:

CAUSE:

PLANT CONDITIONS:

EFFECTS:

REFERENCES:

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Generator Insulation Breakdown

NUMBER: EG07

TYPE: Discrete, Variable (0 - 100% scale on Generator Core Monitor Recorder)

CAUSE: Insulation Degradation

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in main generator insulation breakdown due to insulation degradation.

Generator insulation is monitored via generator core monitor system. Generator core monitor meter and recorder indication will decrease dependent on severity selected resulting in remote alarm actuation on CRP 9-23 and "Gen Core Mon" alarm actuation on CRP 9-8, at 50% +/-2% of scale. During normal operation the generator core monitor reads approximately 90% of scale or greater. Plant response is dependent on operator action.

Removal of the malfunction will require re-initialization of the Simulator.

REFERENCES: 104-00826, Stator Cooling Water System
104-01100, Generator Core Monitor System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Generator Main Seal Oil Pump Trip

NUMBER: EG08

TYPE: Discrete

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS:

This malfunction will cause the generator main seal oil pump to trip due to overload trip.

The emergency seal oil pump will auto-start at 80 psig as seal oil pressure decreases. "Hydrogen and Stator Cooling Panel Trouble" alarm will actuate on CRP 9-7.

Main generator hydrogen pressure will remain constant. Generator hydrogen seal pressure differential will be less due to smaller capacity of emergency seal oil pump of 54 gpm versus main seal oil capacity of 66 gpm. Hydrogen losses in sealing oil will be slightly less than normal. With the emergency pumps in operation, oil enters the pump suction through valve H-14 which connects with the bearing drain. The vacuum tank becomes inoperative with ESOP in operation, and float valve H-1C closes so that no oil enters the tank.

The oil pumped to the shaft seals from the emergency oil pump passes through regulating valve H-19. This valve operates to control the seal oil pressure the same as in normal operation of the shaft sealing system.

Removal of the malfunction will allow restart of the generator main seal oil pump.

REFERENCES: GEK 5585, Vol II, Section 29
RP 2161, Generator Hydrogen/Seal Oil System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Generator Hydrogen Leak

NUMBER: EG09

TYPE: Discrete, Variable (0 - 100% Severity = 0 - 10 psig/minute at 45 psig)

CAUSE: Generator Casing Break

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will result in a generator hydrogen leak due to a generator casing break. The magnitude of the leak will depend on the severity selected. Indicated hydrogen pressure will decrease and hydrogen purity will slowly decrease as indicated on CRP 9-7.

As the hydrogen pressure decreases a loss of generator cooling will occur. The rate of temperature increase and amount will depend on the amount of hydrogen lost and generator loading.

Removal of the malfunction will return machine gas pressure to normal.

REFERENCES: GEK 5585
OP 2161

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Emergency Seal Oil Pump Trip

NUMBER: EG10

TYPE: Discrete

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating at Power, Main Generator On-Line

EFFECTS: This malfunction will cause the emergency seal oil pump to trip due to overload, if ESOP is running and/or prevent the auto/manual start if not running. Plant response will be minimal except for common alarm "Hyd and Stator Clg Panel Trouble" on CRP 9-7. In order to ensure an oil supply to the shaft seals with both the main and emergency seal oil pumps inoperative, a secondary emergency seal supply system is provided, which is effective at hydrogen pressure up to about 8 psi. This system consists of check valve H-17 connected between the bearing and seal oil headers.

Removal of the malfunction will return emergency seal oil pump to normal.

REFERENCES: GEK 5585, Vol. II Sect 29
RP 2161 Generator Hydrogen/Seal Oil and Generator Core Monitor System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

__TITLE:__ Loss of Generator Field Breaker

__NUMBER:__ EG11

__TYPE:__ Discrete

__CAUSE:__ Short Circuit Acrcss Contact 9-9C (41/Aux) on the 286/B-B Relay

__PLANT CONDITIONS:__

Reactor Operating at Power, Main Generator On-Line

__EFFECTS:__ .. This malfunction will cause the main generator field breaker to trip. Main generator field voltage and field amperage will drop to zero resulting in turbine trip (MTS-1) on loss of field. The main generator's output breakers 1T and 81-1T trip on generator lockout. The plant response will be similar to Malf EG01 except the field breaker tripped before the generator. All instrumentation and annunciators will respond as required.

Removal of malfunction will allow reclosure of the field breaker.

__REFERENCES:__ GEK 5585, Vol. II
CWD B-191301, sh. 203

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Pump (A,B,C) Trip

NUMBER: FW08

TYPE: Generic (A,B,C)

CAUSE: Activation of Instantaneous Overcurrent Device (50)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected feedwater pump(s) to trip due to activation of overcurrent device (50). Reactor feed pump trip aununciator will actuate. Feedwater pumps discharge header pressure and flow indication on CRP 9-6, 9-5 will decrease. Reactor water level will start decreasing. The feedwater control valves will ramp open trying to maintain reactor water level. The standby reactor feedwater pump, if in auto, will automatically start. As the standby pump comes up to pressure it will begin feeding at an increased rate and reactor water level will begin to recover. If reactor water reaches its low trip setpoint, a reactor scram will occur.

Attempting to restart the pump will result in the breaker closing and tripping immediately.

Removal of the malfunction will allow normal operating of the selected feedwater pump.

REFERENCES: P+ID G-191157
CWD B-191301 sh. 550-552
OP 2172, Feedwater System Operation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Air to Feedwater Regulating Valve (A,B)

NUMBER: FW09

TYPE: Generic (A,B)

CAUSE: Instrument Air Line Rupture

PLANT CONDITIONS:

Reactor Operating at Power, 10-100%

EFFECTS: This malfunction will cause the selected feedwater reg. valve air line, between the valve controller and valve actuator, to rupture. This results in a complete loss of air to the valve. The level control valve will remain in its present position and will not respond to any signals from the control system. The loss of air pressure will actuate the feedwater valve control signal failure annunciator on CRP 9-5. The valve reset lockup lamp will illuminate. Pressing the valve lockup reset pushbutton will extinguish the lamp but will have no effect on the affected feedwater reg. valve. The lamp will come back on when the pushbutton is released.

Level transients will be slightly greater than normal, since the unaffected feed reg. valve will have to move twice as much to get the required flow. If the flow demand exceeds the combined capacity of the affected valve at its maximum position, level will begin to decrease.

If flow demand decreases below the capacity of the affected valve at its present position, the unaffected valve will complete close and level will begin to increase.

Removal of the malfunction will allow the feed reg. valve to be reset and returned to normal operation.

REFERENCES: P+ID G-191157
OP 2172
CWD B-191301, Sh. 502, 504, 507

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Regulating Valve (A,B) Controller Failure

NUMBER: FW10

TYPE: Generic (A,B), Variable (100% = Valve Full Open Demand)

CAUSE: Failure of Control Auto Output Signal

PLANT CONDITIONS:

Reactor Operating at Power, 10-100%

EFFECTS: This malfunction will cause the selected feedwater regulating valve controller auto output signal too fail too the instructor specified value. The Feed Reg. Valve will respond by moving to the demanded position.

If the selected valve opens from its steady state position the other valve will close down as feed flow and reactor level increase to new stable values.

If the selected valve closes from its steady state position, the other valve will open in response to the decreased feel flow and reactor level. Since the capacity of each feedwater level control valve is 55% of rated flow, the reactor will not get sufficient flow to maintain level. Reactor level will continue to decrease until a scram occurs due to low level.

Placing the selected controller in manual will block the auto signal and allow the operator to position the feedwater reg. valve.

Removal of the malfunction will restore the selected controller auto output signal to normal operation.

REFERENCES: P+ID G-191157
OP 2172
GEK 32431

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Master Level Controller Failure

NUMBER: FW11

TYPE: Discrete, Variable (0-100% of Demand)

CAUSE: Failure of Controller Auto Output Signal

PLANT CONDITIONS:

Reactor Operating at Power, 10-100%

EFFECTS: The malfunction will cause the feedwater master level controller auto output signal to fail to the instructor specified value. The feed reg. valves, if in auto, will respond by moving to the demanded position. If an individual feed reg. valve is in manual, that feed reg. valve will not be affected.

If maximum demand from the master controller is selected, reactor water level will increase rapidly, the reactor feed pumps and main turbine will trip due to high reactor level. The reactor will scram due to turbine stop valve closure.

If minimum demand from the master controller is selected. Reactor water level will decrease as a function of steam flow/feed flow mismatch. The feed pump minimum flow valves will open. Reactor recirculation system runback on low total feed flow without any operator action, reactor level will decrease to the low level scram setpoint.

Placing the feedwater master level controller in manual will block the auto signal and allow the operator to position the feed reg. valves manually via the master level controller.

Removal of the malfunction will restore the master level controller auto output signal to normal operation.

<u>REFERENCES</u>: OP 2172
GEK 32431

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feed Pump Minimum Flow Valve (A,B,C) Fails Closed

NUMBER: FW12

TYPE: Generic (A,B,C)

CAUSE: Selected I/P Converter Output Signal Fails

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the selected feed pump minimum flow valve, if open to close. If the minimum flow valve is already closed it will not open. Minimum flow valve closed indicating light on CRP 9-6 will illuminate.

If the affected feedwater pump is started and running for greater than approximately 8 seconds and flow is less than 300,000 LB/HR. The feed pump will trip on a low flow signal (2 seconds). The affected reactor feed pump Flo/oil PR Lo/trip annunciator on CRP 9-6 will actuate. All other associated annunciators will actuate when appropriate. See feedwater pump trip effects (FW08).

Removal of malfunction will restore the affected I/P converter to normal operation.

REFERENCES: P+ID G-191157
CWD B-191301 sh. 559
OP 2172

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Startup Feedwater Regulator Controller Failure

NUMBER: FW13

TYPE: Discrete, Variable (100% = Valve Fully Open)

CAUSE: Failure of Controller Auto Output Signal

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the startup feedwater regulator controller auto output signal to fail to the instructor specified value.

If this malfunction is inserted at low power or shutdown conditions when the 10% feed valve is controlling level, the level will respond according to the deviation between feed flow to the vessel and losses from the vessel.

Placing the startup feedwater regulator controller in manual will block the auto signal and allow the operator to manually maintain reactor water level.

Removal of the malfunction will restore the startup feedwater reg controller auto output signal to normal operation.

REFERENCES: P+ID G-191157
OP 2172
GEK-32431

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Failure of Steam Flow Summer

NUMBER: FW14

TYPE: Discrete, Variable (0 - 100% of Scale)

CAUSE: Steam Flow Summer Output Signal Fails

PLANT CONDITIONS:

Reactor Operating at Power, Feedwater in 3 Element Control

EFFECTS: This malfunction will cause the steam flow summer to fail to the instructor specified value. A steam flow/feed flow mismatch will occur.

If the total steam flow is less than total feed flow, a flow error signal is sent to the master level controller, the feed reg valves will respond and begin to close.

Reactor water level will decrease until the reactor water level signal input to the master controller restores level to lower than its original value.

If the total steam flow is greater than total feed flow, the feed reg valves will respond and begin to open. Reactor water level will increase until the reactor water level signal input to the master controller restores level to higher than its original value.

If at any time reactor water level reaches its low level setpoint a reactor scram will occur. If it reaches its high level setpoint, a turbine trip will occur.

Removal of malfunction will restore the steam flow summer output signal too normal operating value.

<u>REFERENCES</u>: GEK 32431
CWD B-191301 sh. 503-504

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Failure of Feedwater Flow Summer

NUMBER: FW15

TYPE: Discrete, Variable (0-100% of Scale)

CAUSE: Feedwater Flow Summer Output Signal Fails

PLANT CONDITIONS:

Reactor Operating at Power, Feedwater in 3 Element Control

EFFECTS: The malfunction will cause the feedwater flow summer to fail to the instructor specified value. A steam flow/feed flow mismatch will occur.

If total feed flow is less than total steam flow, a flow error signal is sent to the master level controller, the feed reg. valves will respond and begin to open. Reactor water level will increase until the reactor water level signal input to the master controller restores level to higher than its original value.

If total feed flow is greater than total steam flow, the feed reg. valves will respond and begin to close. Reactor water level will decrease until the reactor water level signal input to the master controller restores level to lower than its original value.

Removal of the malfunction will restore the feedwater flow summer signal to normal operating value.

REFERENCES: GEK 32341
CWD B-191301, Sh. 506

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Control Level Signal (A,B) Failure

NUMBER: FW16

TYPE: Generic (A,B), Variable (0-100% of Scale)

CAUSE: Selected Level Transmitter (52A,B) Output Signal Fails

PLANT CONDITIONS:

Reactor Operating at Power, Feedwater In 3 Element Control

EFFECTS: This malfunction will cause the selected level transmitter to fail to the instructor specified value. The affected level indicator and recorder on CRP 9-5 will reflect the severity selected.

If the level signal selected is greater than the actual reactor level, the feed reg. valves will respond and begin to close. Reactor water level will begin to decrease. If reactor water level reaches its low level setpoint a reactor scram will occur.

If the level signal selected is less than the actual reactor level, the feed reg. valves will respond and begin to open. Reactor water level will begin to increase. If reactor water level reaches its high level setpoint the reactor feedwater pumps and turbine will trip. The reactor will scram due to turbine stop valve closure.

The operator can select the unaffected alternate level channel and allow reactor water level to be restored. Removal of the malfunction will restore the affected level signal to normal operating value.

REFERENCES: GEK 32431
CWD B-191301, Sh. 505

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: High Pressure Feedwater Heater (1A, 2A, 1B, 2B) Tube Leak

NUMBER: FW18

TYPE: Generic (1A, 2A, 1B, 2B), Variable (100% = 150% Drain Capacity)

CAUSE: Tube Failure

PLANT CONDITIONS:

Reactor Operating at Power, Turbine On-Line

EFFECTS: This malfunction will cause the selected high pressure feedwater heater tubes to leak at the severity selected. This results in a loss of feed/condensate to the affected heater shell. As the shell level increases the normal level control valve will open. If the leak rate is higher than the drain rate, level will increase until the emergency level control valve opens to the condenser. The high pressure feedwater heater high level annunciator will actuate. If the affected heater reaches the Hi-Hi level setpoint, the associated extraction non-return valve will close and the extraction steam dump valve will open.

The flow lost through the heater leak will decrease the flow being delivered to the reactor vessel. Reactor level will decrease slightly as the feed reg. valves open to stabilize level.

Feedwater heater temperature will decrease resulting in a decrease of feedwater temperature at the heater outlet and will be transmitted through the string. This results in an increase of reactor power but a loss of plant efficiency.

Removal of the malfunction will restore the high pressure feedwater heater(s) tubes to normal.

<u>REFERENCES</u>: P+ID G-191157, 191158
RP 2172

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of High Pressure Heater (A,B) String

NUMBER: FW20

TYPE: Generic (A,B)

CAUSE: High Pressure Heater Train Inlet Valve Switch (9-6-51, 9-7-52) Close Contacts (3-3T) Weld Close

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the selected high pressure heater train inlet valve to close. The selected valve closed indicating light on CRP 9-6 will illuminate. Reactor feedwater inlet temperature will decrease causing an increase in core inlet subcooling. Due to the negative moderator temperature coefficient, and increase in core power results. Reactor water level will be maintained by feedwater reg. valves. All associated annunciators will actuate when appropriate.

Opening of the high pressure heater bypass will return the FW system pressures and flow to normal.

Removal of the malfunction will restore the closing contacts to normal and allow the affected high pressure heater string to be returned to normal operation.

REFERENCES: P+ID G-191157
CWD B-191301, Sh. 509-510
OP 2172

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Line Break Outside Containment

NUMBER: FW21

TYPE: Discrete, Variable (100% = Complete Pipe Degradation, 10 Inch Pipe)

CAUSE: High Pressure Heater Outlet Cross-connect Pipe Failure

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause the high pressure feedwater heater outlet cross-connect pipe to break to the instructor specified value.

At 100% severity and normal feed pressure, the rupture will cause the feedwater header pressure to drop rapidly too less than reactor pressure. The containment isolation check valves will prevent any reverse flow from the reactor vessel. The loss of feed flow to the vessel will cause reactor water level to decrease, and the level control system will attempt to restore level by opening the feed regulating valves. As flow through the feedwater pumps increase, both suction and discharge pressure will decrease as the pumps go into runout. As suction pressure decreases, the low suction pressure annunciator will actuate and the feedwater pumps will eventually trip.

The reactor will scram on reactor low level. Automatic activation of the ECCS systems will restore level in the reactor vessel.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191157, 191167
FSAR Section 14.5.4.3

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Pump (A,B,C) Fails to Auto Start

NUMBER: FW22

TYPE: Generic (A,B,C)

CAUSE: Auto Start Contacts (8-8T) on Switch SW-9-6-9, 10,11 Fail Open

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected feedwater pump auto start contacts to fail open. No immediate effects will be observed. If the feedwater pump is running and the malfunction is activated. No effects will be observed. If a running feedwater pump trips and its switch is in auto and the malfunction is active, the selected feedwater pump will not auto start. Feedwater system flow, pressure and reactor water level will reflect the inability of the standby pump to start.

The affected feedwater pump can be started manually and restore reactor water level to normal.

Removal of malfunction will restore the auto start contacts to normal.

REFERENCES: P+ID G-191157
CWD B-191301, Sh. 550-552
OP 2172

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Line (A,B) Break Inside Containment

NUMBER: FW23

TYPE: Discrete, Varible (100% = Complete Pipe Degradation)

CAUSE: Pipe Rupture Down Stream of Isolation Check Valves and Manual Isolation Valves V29A/V29B.

PLANT CONDITIONS:

EFFECTS: This malfunction will cause the selected feedwater line to break, to the instructor specified value, resulting in an unisolable leak from the reactor pressure vessel through the selected feedwater header.

The effects of the feedwater system will be similar to those observed in FW21 malfunction. The loss of water from the reactor vessel will cause containment pressure and temperature to rise. As the water level in the vessel downcomer decreases below the level of the feedwater spargers, the flow from the vessel becomes steam causing the rate of mass loss from the vessel to noticeably decrease and the rate of vessel depressurization to noticeably increase.

LPCI and core spray will initiate to recover level and prevent core damage, PCIS group isolations will occur.

Removal of malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191167
FSAR, Section 14

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Feedwater Line (A,B) Break Inside Containment

NUMBER: FW27

TYPE: Generic (A,B)

CAUSE: Feedwater Line (A,B) Break Inside Drywell Between Drywell Wall and Check Valve

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS:

This malfunction will cause the selected feedwater line to break. The break will occur inside the drywell between the drywell wall and associated check valve, resulting in a semi-isolable (leak) break.

The feedwater check valve will function as designed to limit loss of vessel inventory to the break. The loss of water from the feed system will cause containment pressure and temperature to increase. All functions associated with hi drywell pressure will occur.

Removal of this malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191167
FSAR, Section 14

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Turbine Trip

NUMBER: HP01

TYPE: Discrete

CAUSE: Failure of Contact K40(2-8) `Rx High Water Level'

PLANT CONDITIONS:

HPCI System Operating

EFFECTS: This malfunction will cause the HPCI turbine to trip. HPCI turbine stop valve will close. HPCI turbine will begin to coast down, pump pressure and system flow will approach static conditions. Turbine trip annunciators will actuate. If the system was supplying water to the reactor, level will begin to decrease at a rate dependent upon other water supply systems in operation.

Removal of malfunction will restore PS-23-97A to normal operation and allow HPCI system to operate normally.

REFERENCES: GEK 9613
CWD B-191301 Sh. 1450

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Failure to Auto Start

NUMBER: HP02

TYPE: Discrete

CAUSE: 23A-K1, K2, K3, K4 Relay Failure Inhibits HPCI Auto Start

PLANT CONDITIONS:

Conditions Requiring HPCI System Auto Start

EFFECTS: This malfunction will result in the failure of the HPCI system to auto start on an initiation signal. Reactor parameters will reflect the loss of this system and will be dependent on remaining equipment in operation. The operator will have the capability of manually starting the HPCI system and inject into the vessel.

Removal of the malfunction will return relays 23A-K1, K2, K3, K4 to normal operation.

REFERENCES: GEK - 9613
CWD B-191301, Sh. 1449

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Inadvertent Initiation

NUMBER: HP03

TYPE: Discrete

CAUSE: Electrical Short in the Reactor Low Level Sensing Circuit, Resulting in Energizing K1 and K2 Relays

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause the HPCI system to automatically start. HPCI pump discharge pressure will begin to increase. Water will begin to be injected as soon as pump discharge pressure overcomes reactor pressure and vessel level will begin to increase. This will result in a reactivity addition which may cause a reactor scram due to high flux. The feedwater control system will compensate for the increase in reactor water level.

If a scram does not occur, level will return to approximately the original value with a lower feedwater temperature and higher reactor flux levels.

Removal of the malfunction will restore HPCI to normal operation.

REFERENCES: GEK 9613
CWD B-191301 sh. 1449

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Flow Controller Failure

NUMBER: HP04

TYPE: Variable (0-100% of Range)

CAUSE: Controller Output Signal Failure in Auto Mode

PLANT CONDITIONS:

HPCI System Operating

EFFECTS: This malfunction will cause the HPCI flow controller output signal to fail to the severity selected. If the HPCI flow control signal fails low, the HPCI turbine speed will decrease to its low speed limit if operating in auto. System flow and pressure will reflect the turbine speed decrease. If the malfunction is activated prior to a system start, the HPCI turbine will increase to its low speed stop and remain at that speed.

If the HPCI flow control signal fails high, the HPCI turbine speed will increase to its high speed limit if operating in auto. System flow and pressure will reflect the turbine speed increase. If this malfunction is activated before the system starts, the turbine will increase in speed and trip due to overspeed.

The operator may place the controller in manual and adjust the flow controller to the desired value.

Removal of malfunction will restore the controller output to normal.

REFERENCES: P+ID G-191169
GEK-9613
CWD B-191301, Sh. 1452

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Inadvertent Isolation

NUMBER: HP05

TYPE: Discrete

CAUSE: Failure of DPIS 23-76

PLANT CONDITIONS:

HPCI System in Operation

EFFECTS: This malfunction will result in the loss of steam supply to the HPCI Turbine due to closure of the inboard and outboard steam isolation valves. The HPCI steam line Hi Diff pressure annunciator will actuate. HPCI turbine will trip as a result of systems auto isolation. HPCI turbine speed will decrease as a function of the decreasing steam supply pressure and turbine coast down characteristics. HPCI pump discharge pressure and system flow will decrease correspondingly.

Removal of the malfunction will restore the HPCI system to normal operation.

REFERENCES: GEK 9613
CWD B-191301 sh. 1455

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Exhaust Diaphram Failure

NUMBER: HP06

TYPE: Discrete

CAUSE: Exhaust Diaphram Rupture

PLANT CONDITIONS:

HPCI System Operating

EFFECTS: This malfunction will cause the HPCI exhaust diaphram to rupture resulting in steam being released to the torus area atmosphere. Radiation monitoring system will detect an increase in activity. HPCI flow controller will maintain system flow constant in spite of increased HPCI turbine steam flow. When the steam leak detecting system high temperature setpoint is reached, the HPCI system will automatically isolate. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. Turbine exhaust pressure indicated on CRP 9-3 will decrease. All associated alarms will actuate when appropriate.

Removal of the malfunction will return the HPCI system to normal operation.

REFERENCES: GEK-9613
P+ID G-191169

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Injection Valve (HPCI-19) Fails to Auto Open

NUMBER: HP07

TYPE: Discrete

CAUSE: HPCI-19 Auto Open Contacts (5-6, 9-10) Fail Open

PLANT CONDITIONS:

HPCI System Auto Start

EFFECTS: This malfunction will allow the HPCI system equipment to operate as normal on an initiation signal with the exception of valve HPCI-19 remaining closed. This will prevent water from being admitted to the reactor. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. The operator can manually open HPCI-19 from 9-3 and restore flow to vessel.

Removal of the malfunction will restore the HPCI injection valves response to an auto initiation signal.

REFERENCES: P+ID G191169, Sh. 2
GEK-9613
CWD B-191301, Sh. 1440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: HPCI Speed Control Fails (High/Low)

NUMBER: HP08

TYPE: Discrete; A=High, B=Low

CAUSE: Control Signal From Speed Governor Fails to Selected Maximum/Minimum Output in "Auto" Mode

PLANT CONDITIONS:

HPCI System Operating

EFFECTS: This malfunction will cause the HPCI speed control to fail:

HPCI speed control fails high: This malfunction will cause the HPCI turbine speed to increase to its high speed limit, as long as the speed controller is in "Auto" during HPCI operation. HPCI system pressure and flow will reflect the turbine speed increase. If the turbine is started after the malfunction is activated, then turbine speed will increase to the overspeed trip setpoint of 5000 RPM. The operator may place the HPCI speed controller in manual and adjust turbine speed to the desired value.

HPCI speed control fails low: This malfunction will cause the HPCI turbine speed to decrease to its low speed limit (2100 RPM) as long as the speed controller is in "Auto" during HPCI operation.

HPCI system pressure and flow will reflect the turbine speed decrease. If HPCI is injecting to the vessel, flow may stop completely, dependent upon reactor pressure. If the turbine is started after the malfunction is activated then turbine speed will increase to low speed limit and remain there. The operator may place the HPCI speed controller in manual and adjust turbine speed to the desired value.

Removal of malfunction will return the HPCI speed governor output to normal.

<u>REFERENCES</u>: GEK-9613
P+ID G-191169
CWD B-191301, Sh. 1452

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

<u>TITLE</u>: HPCI Steam Line Leak

<u>NUMBER</u>: HP09

<u>TYPE</u>: Discrete, Variable (100% = 4 inch pipe)

<u>CAUSE</u>: Piping Failure on HPCI Steam Supply Line in the HPCI Room up stream of the HPCI-14 valve.

<u>PLANT CONDITIONS</u>:

HPCI System in Service

<u>EFFECTS</u>: This malfunction will result in the isolation of the HPCI system steam supply. Both the inboard and outboard steam isolation valves will close simultaneously when the HPCI steam line area temperature reaches 200^{0}F or Hi flow at 300%.

Prior to closure of the HPCI steam supply valves, an annunciator will actuate at 185^{0}F, increasing, warning the operator of a possible steam line leak and/or break. The appropriate leak detection parameters will indicate increasing area temperature in the HPCI room. Upon isolation turbine trip will be initiated. HPCI turbine speed will decrease as function of decreasing turbine steam supply pressure, and HPCI turbine coastdown characteristics HPCI pump discharge pressure and system flow will decrease accordingly.

Removal of the malfunction will allow the HPCI system returned to normal operation.

<u>REFERENCES</u>: GEK-9613
CWD B-191301 sh. 1450
P+ID G-191169

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Station Air Compressor (A,B,C,D) Trip

NUMBER: IA01

TYPE: Generic (A,B,C,D)

CAUSE: Activation of Thermal Relay Device (49)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected station air compressor, if running or when started, to trip due to activation of the thermal overload device. The compressor running/stop indicating lights on PNL 9-6 will extinguish. System pressure will decrease and the station air compressors that are in lag position will automatically start at approximately 95 PSI.

Removal of the malfunction will reset the overload trip device and allow normal operation of the affected station air compressor.

REFERENCES: CWD B-191301, Sh. 580-582
OP 2190
P+ID G-191160

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Instrument Air Header Leak

NUMBER: IA02

TYPE: Discrete, Variable (100% = 2 square inches in dryer)

CAUSE: Air Dryer "A" Leak

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause instrument air header "A", downstream of the Instrument Air dryer, to leak at the severity selected. At 100% severity the Instrument Air header will immediately depressurize and the Instrument Air receiver will rapidly depressurize. The standby air compressors will auto start and the Service Air header will isolate. All components supplied by instrument air will go to their fail positions as the pressure decreases. Plant shutdown will occur as the control rod drive scram air header pressure decreases, resulting in reactor scram and associated turbine trip. All associated annunciators will actuate when appropriate.

At small severities, Instrument Air header pressure will decrease much slower and may stabilize. At a lower operating pressure depending upon the severity and the number of compressors in service.

Removal of the malfunction returns Instrument Air to normal.

REFERENCES: P+ID G-191160
OP 2190

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Primary Containment Instrument Air Header Leak

NUMBER: IA03

TYPE: Discrete, Variable (100% = 50% Pipe Diameter, 2 inch pipe)

CAUSE: Failure on Instrument Air Ring Header in Drywell

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the primary containment instrument air ring header to leak at the severity selected. At 100% severity and N_2 flange removed via IDA IAR08, the instrument air header inside the drywell will immediately depressurize. The primary containment atmosphere compressor receiver pressure will rapidly decrease and drywell pressure will increase. When receiver pressure decreases to approximately 95 psig the primary containment atmosphere compressor will auto start, if control switch is in auto.

If the drywell pressure reaches 2.5 psig, containment isolation and a reactor scram will occur. All components supplied by the primary containment instrument air header will go too their fail position. All associated annunciators will actuate when appropriate.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191160, sh. 3,4
CWD B-191301, sh. 583
OP 2191

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Scram Air

NUMBER: IA04

TYPE: Discrete, Variable (100% = Scram Pipe Header Diameter, 1.5 inch pipe)

CAUSE: Scram Valve Pilot Air Header Pipe Failure, Located Downstream of Air Filter

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the scram valve pilot air header to leak at the severity selected. At 100% severity, the scram valve pilot air header pressure will decrease rapidly and at approximately 60 psig, the scram pilot header Hi/Lo annunciator on CRP 9-5 will actuate. As pressure continues to decrease, scram valves will open and control rods will begin to individually scram.

At small severities, scram air header pressure will decrease much slower.

Removal of the malfunction will return Scram Air Header to normal.

REFERENCES: P+ID G-191160, 119170
OP 2111

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Air Header Leak

NUMBER: IA05

TYPE: Discrete, Variable (100% = Pipe Diameter Downstream of PCV-1 (4 inch pipe))

CAUSE: Piping Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the service air header to leak at the severity selected. If 100% severity is selected the service air header will rapidly decrease. The instrument air header pressure will start to decrease. The lag compressor will auto start and the instrument air receiver low pressure annunciator will actuate. When station air header pressure decreases to approximately 85 psi and decreasing, PCV-1 will close and isolate the break. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore the service Air Header to normal.

REFERENCES: P+ID G-191160, sh. 5
OP 2190

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Water Pump (A,B,C) Trip

NUMBER: MC01

TYPE: Generic (A,B,C)

CAUSE: Overcurrent Relay Trip (50)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS:

This malfunction will cause the selected circulating water pump to trip. Motor amps will spike then decrease immediately to zero. Decrease circulating water flow will cause condenser temperature to increase and condenser vacuum to decrease resulting in a reduction in plant efficiency and generator megawatt output.

The magnitude of the vacuum decrease is dependent upon the power level and the combination of circ water pump failures. If the condenser vacuum continues to decrease, the turbine will trip causing generator to trip.

Removal of the malfunction will allow restart of affected circulating water pump.

REFERENCES: CWD B-191301, sh. 410-412
OP 2180

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Water Booster Pump (A,B,C) Trip

NUMBER: MC02

TYPE: Generic (A,B,C)

CAUSE: Overcurrent Relay Trip (50)

PLANT CONDITIONS:

Closed Cycle Operation

EFFECTS: This malfunction will cause the circ water booster pump to immediately trip. Motor amps will spike then immediately decrease to zero. Booster pump discharge valve will close. Flow to cooling towers will decrease CW booster pump trip annunciator will actuate.

With the loss of CW booster pump the intake structure level will begin to decrease and the discharge basin level will increase. Depending on plant condition, starting an alternate CW booster pump and/or manipulation of discharge basin bypass gates will be necessary. Associated annunciators will actuate when appropriate.

Removal of the malfunction will restore affected CW booster pump to normal operation.

REFERENCES: P+ID G-191160, sh. 5
OP 2190

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Water Traveling Screen (A,B,C) Fouling

NUMBER: MC03

TYPE: Generic (A,B,C), Variable (100% Circ Water Pump Trip)

CAUSE: Debris in Circulating Water Intake

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the circulating water traveling screen to foul to the desired severity. Screenwash will have no effect on the rate at which the screen differential level increases. The CW pump(s) affected will trip on low level resulting in reduction of plant efficiency and possible loss of unit due to condenser vacuum trips.

After the CW pump trips, the diff level will slowly decrease back toward zero. If the CW pump is restarted, the differential level will again increase as a function of flow out of the suction bay. Associated alarms will actuate when appropriate.

Removal of the malfunction will restore the traveling screen back to normal operation.

REFERENCES: P+ID G-191166
CWD B-191301, sh. 410-412

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Water Expansion Joint (A,B,C,D) Leak

NUMBER: MC04

TYPE: Generic (A,B,C,D), Variable (100% = 100,000 gpm)

CAUSE: Circulating Water Inlet to Condenser Expansion Joint Fails

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in a loss of circulating water to the turbine building floor at the selected severity. Total system flow will increase and header pressure will decrease reflective of severity selected. This will result in a lower condenser differential pressure and decrease in condenser efficiency. Generator load will decrease as vacuum decrease.

If the leak is not isolated the turbine building floor sump alarm will actuate at some point dependent upon malfunction severity. The operator can isolate the leak by isolating the affected water box.

Removal of the malfunction will restore circulation system to normal.

REFERENCES: P+ID G-191166

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Pump Discharge Valve (A,B,C) Fails to Close

NUMBER: *MC05

TYPE: Generic (A,B,C)

CAUSE: Valve Stem Shears

PLANT CONDITIONS:

Any Plant Conditions

EFFECTS: This malfunction will cause the selected Circ. water pump discharge valve not to close from its full open position when the affected CW pump stop signal is given. The CW pump discharge valve will indicate normally. The affected CW pump will stop as the discharge valve indication closes.

If other Circ. water pumps are operating, their amps will decrease when the pump trips due to short cycling of circ water then the open valve(s).

If the pump is restarted with the discharge valve failed open, the pump will trip on excessive starting current. The amount of vacuum loss from the main condenser due to the amp trip and the short cycling of circ water will be dependent on plans operative conditions.

REFERENCES: CWD G-191301, Sh. 410-418

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Circulating Water Booster Pump Discharge Valve (A,B,C) Fails to Continue Opening After Second Start Signal

NUMBER: MC06

TYPE: Generic (A,B,C)

CAUSE: Valve Binding Results in Thermal Overload (49)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: Valve operation will appear normal and the affected CWB pump will start at 20% valve position. When the second start signal if given, indication for the valve will be lost when the thermal overload contacts in the control power circuit open. The affected CWB pump amps will remain constant. Valve position will remain at 20%.

Removal of the malfunction will restore the selected valve to normal operation, and reset the thermal overload.

REFERENCES: CWD B-191301, Sh. 419-421

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condenser (A,B,C,D) Tube Leak

NUMBER: MC07

TYPE: Generic (A,B,C,D), Variable (100% = 10 gpm)

CAUSE: Tube Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected area condenser tubes to leak. The condenser area hotwell conductivity will increase as monitored on CRP 9-23. The conductivity within the hotwell of the selected area leak will increase first and then be followed by the conductivity increase of the other hotwell monitor as the fluid is mixed together, high conductivity annunciator on CRP 9-6 may actuate.

All associated condensate, feedwater, and RWCU flows will correctly propagate the conductivity effects to associated tanks and vessels as appropriate. Associated conductivity and demineralizer alarms will be actuated.

Removal of the malfunction will stop the leak and allow cleanup of the system by normal means.

REFERENCES: P+ID G-191157, 191164

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condenser Air In-Leakage

NUMBER: MC08

TYPE: Discrete, Variable (100% = 1000 SCFM)

CAUSE: Condenser "A" Casing Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in air leakage into condenser "A" at the selected severity. An increase in air leakage will result in an overall loss of plant efficiency as condenser vacuum decreases, the turbine generator load will decrease. The vacuum in condenser "B" will also begin decreasing since the condensers are connected by an equalizing line. The increased air leakage will result in a higher offgas flow rate.

For low severities, condenser vacuum will stabilize at a lower value. As the severity is increased the SJAES will not be able to handle the additional load. Condenser vacuum will continue to decrease to the low vacuum alarm trip setpoint, resulting in a turbine trip and subsequent reactor scram. All associated annunciators will actuate when appropriate. Removal of malfunction will stop the leak, however, condenser vacuum recovery will be dependent upon the severity selected and plant status.

REFERENCES: P+ID G-19115

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condenser Tube (A,B,C,D) Rupture

NUMBER: MC09

TYPE: Generic (A,B,C,D), Variable (100% = 1000 GPM)

CAUSE: Tube Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected area tubes to rupture. The magnitude of the rupture will be determined by the severity selected. The affected area tube rupture will be evident by a rapid increase in conductivity in one area of the condenser and will rapidly spread throughout.

The intrusion will promptly propagate through the condensate and feedwater systems and the reactor vessel. RWCU system influent and effluent conductivity will increase. Rapid depletion of all in-service resin beds in aforementioned flow paths will result. All associated condensate, feedwater and RWCU flows will correctly propagate the conductivity effects to associated tanks and vessels as appropriate. Hotwell level and reject flow will both increase according to severity. If hotwell level is increased to the point of covering the condenser tubes, condenser vacuum will decrease, thus reducing plant efficiency.

Isolation of the water box will result in stoppage of all water leakage. At lesser severities, plant recovery to full power will be possible by deactivating the malfunction, and cleanup of system by normal means.

REFERENCES: P+ID G-191157, 191164

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Condenser Tube Sheet (A,B,C,D) Fouling

NUMBER: MC10

TYPE: Generic (A,B,C,D), Variable (100% = Complete Blockage)

CAUSE: Marine Growth

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause circulating water flow through the selected condenser section to stop when 100% severity is chosen. System pressure and circulating water pumps amps would increase slightly. Condenser vacuum will decrease. Generator load will decrease as a function of condenser backpressure. If reactor power is not decreased a turbine trip will result.

Lesser severities will reduce circulating water flow by the specified severity and pump response will be in accordance with the appropriate performance curves.

Removal of the malfunction will restore the affected water box tube sheet to normal.

REFERENCES: P+ID G-191166
OP 2180

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SJAE Pressure Regulator (PCV-1) Failure

NUMBER: MC11

TYPE: Discrete, Variable (0 - 100% Valve Position)

CAUSE: Mechanical Failure of the SJAE Pressure Regulator

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the SJAE pressure regulator to fail to the instructor specified value. Pressure indication and SJAE stream pressure Hi/Lo annunciator on CRP 9-6 will actuate.

If SJAE pressure regulator fails open, the SJAE steam supply pressure will increase. This will result in high offgas flow. The net effect will be observable by higher than normal values for SJAE offgas pressure and temperature.

If SJAE pressure regulator fails closed, the SJAE steam supply will decrease. This will remove the necessary driving steam flow/and pressure to operate the SJAE efficiently. This will result in a build up of non-condensable gases in the condenser. If pressure drops to 50 psig, OG-516A and B will isolate and must be reset before they can be opened. Condenser vacuum will begin to decrease with subsequent load decrease. Vacuum will continue to decrease and will result in a turbine trip, and reactor scram.

Removal of malfunction will restore the SJAE pressure regulator to normal and subsequent recovery of condenser vacuum if trip has not occurred.

REFERENCES: P+ID G-191156

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Mechanical Vacuum Pump Suction Valve Controller Failure (AE-FCV-35)

NUMBER: MC12

TYPE: Discrete, Variable (0-100% of Control Range)

CAUSE: Failure of Controller Output (0-100%)

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the mechanical vacuum pump P53-1A suction valve controller to fail at the desired percent of control range (0-100%) due to failure of controller output. For Mechanical vacuum pump suction valve controller failure to 100%, the following major events occur:

A) AE-FCV-35 fails full open
B) Stack Gas flow rate increases
C) Stack gas activity increases
D) Gland seal exhauster pressure changes

On the other hand, for mechanical vacuum pump suction controller failure to 0% the following events occur:

A) AE-FCV-35 fails to full closed position
B) Stack gas flow rate decreases
C) Stack gas activity decays to shutdown values
D) Gland seal exhauster pressure decreases (back pressure increases)

Removal of malfunction will return mechanical vacuum pump suction valve controller (AE-FCV-35) to normal.

REFERENCES: OP 2150
P+ID G-191157

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Mechanical Vacuum Pump T-ip

NUMBER: MC13

TYPE: Discrete

CAUSE: Overload Trip

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the mechanical vacuum pump P53-1A to trip, if running, due to overload. Main condenser vacuum will be lost. Stack gas flow and stack gas activity will decrease. Gland seal exhauster pressure will decrease.

Removal of the malfunction will allow the mechanical vacuum pump to be restarted.

REFERENCES: OP 2150
P+ID G-191157

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Mechanical Vacuum Pump Loss of Control Power

NUMBER: MC14

TYPE: Discrete

CAUSE: Blown Fuse

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will result in the loss of control power to the mechanical vacuum pump due to blown fuse. Mechanical vacuum pump status lamp indication will be lost, the pump will trip, and operation of the control switch will have no effect. AE-FCV-35, mechanical vacuum pump isolation valve will close. Offgas stack flow rate will decrease as well as stack gas activity.

Tripping of the vacuum pump and closure of AE-FCV-35 will result in decay of main condenser vacuum.

Removal of the malfunction will restore control power to mechanical vacuum pump allowing restart as conditions permit.

REFERENCES: OP 2150

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: OG-516 Valve Controller (A,B) Fails Shut

NUMBER: MC15

TYPE: Generic (A,B)

CAUSE: Controller Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected OG-516A(B) valve controller to fail shut due to controller failure. This will cause the steam jet air ejector's butterfly pressure control valve PCV-OG-516A(B) to go fully closed as indicated on CRP-9-50. Actual setpoint will not change. If OG-516 valve controller fails shut for the in-service SJAE, main condenser vacuum will be lost with subsequent plant trip.

Removal of the malfunction will return OG-516 valve controller to normal.

REFERENCES: P+ID G-191157

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Excessive Recombiner Temperature

NUMBER: MC16

TYPE: Discrete

CAUSE: Hydrogen Transient, Catalyst Failure

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will result in the operating offgas recombiner A(B) to experience excessive recombiner temperature due to hydrogen transient. The affected recombiner bed temperatures will increase above 750°F to approximately 1000°F over a 30 minute period. A 125°F rise between recombiner inlet and outlet temperatures will indicate an increase of 1% concentration of hydrogen in the system. Automatic switch over to the standby recombiner will occur at 2% hydrogen by volume. Closing down on OG-516A(B) will slow down the increased hydrogen concentration but will not prevent the temperature increase all appropriate annunciators will actuate as required.

Removal of the malfunction will return the affected recombiner temperatures to normal after 10 minutes.

REFERENCES: OP 2150
Vermont Yankee Off-Gas Modification Description and Operation, Attachment A, Table 5.2 Failure Analysis

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: MSIV (AO-2-80A-D, AO-2-86A-D) Closure Time

NUMBER: MS01

TYPE: Generic, Variable (80A-D = A-D, 86A-D = E-H)

CAUSE: Variable Valve Speed (2-10 Seconds), Timing Valves Out of Adjustment

PLANT CONDITIONS:

Affected MSIVs Open

EFFECTS: This malfunction will result in the selected main steam isolation valve closure time to be more or less than expected normal closure time values. Normal closure of a valve when testing is approximately 3 to 5 seconds. This malfunction will be variable in that the closure time may be adjustable 2-10 sec. affected plant parameters such as steam flow, steam line flow, and resultant transient will reflect the change in closure time. Rx pressure will also reflect the different closure time.
Note: This malfunction does not affect MSIVs in test P.B. mode.

Removal of the malfunction will return the selected MSIV closure time to normal.

REFERENCES: OP 4113, Main and Auxiliary Steam System Surveillance

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: MSIV (AO-2-80A-D, AO-2-86A-D) Fails to Close

NUMBER: MS02

TYPE: Generic (80A-D = A-D, 86A-D = E-H

CAUSE: Mechanically Binds

PLANT CONDITIONS:

MSIVs Open

EFFECTS: This malfunction will prevent the selected main steam isolation valve(s) (AO-2-80 A,B,C,D; AO-2-86 A,B,C,D) from closing after it has opened, due to mechanical binding. Appropriate indications will remain illuminated. Operation of the affected valve control switch and/or test switch will not close the valve. The selected MSIV will not close in response to any automatic or manual signal. The affected main steam line can be isolated with the associated redundant valve if necessary (unless it is also malfunctioning for the same reason).

Removal of the malfunction will allow the selected MSIV to close as conditions require.

REFERENCES: 104-01009 Main Steam System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: MSIV (AO-2-80A-D), AO-2-86A-D) Disc Separation

NUMBER: MS03

TYPE: Generic (80A-D = A-D, 86A-D = E-H)

CAUSE: Mechanical Failure

PLANT CONDITIONS:

Reactor Startup or Power Operating Condition

EFFECTS: This malfunction will cause the selected main steam isolation valve (AO-2-80 A,B,C,D; AO-2-86 A,B,C,D) disc to become separated from its operator due to mechanical failure. The affected MSIV valve indication response will be normal, however, the affected MSIV will not pass any steam beyond the disc. The associated steam line flow indication will show no flow on the affected steam line.

Removal of the malfunction will restore the affected outboard MSIV to normal provided the valve is in the closed position.

REFERENCES: 104-01009 Main Steam System

VERMONT YA"KEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of MSIV Group (A,B) DC Power

NUMBER: MS04

TYPE: Generic (A-Inboard, B-Outboard)

CAUSE: Fuse (F11A, F11B) Blown

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected group of MSIVs (inboard and/or outboard) DC power fuse to be blown. The selected MSIV group indicating lights on CRP 9-3 will extinguish.

Removal of the malfunction will restore blown fuse and return the selected MSIV group indication to normal.

REFERENCES: CWD B-191301, Sh. 1108, 1110, 1113
OP 2113

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Moisture Separator Drain Tank (A,B,C,D) Emergency Level Control Failure

NUMBER: MS05

TYPE: Generic (A-D), Variable (0-100% Valve Controller Output Position

CAUSE: Failure of Level Control (LC23A-D) Output Signal

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the affected emergency level control output signal to fail to the instructor specified value.

As the valve opens, the affected moisture separator emergency drain valve open annunciator on CRP 9-7 will actuate. The normal level control valve will close due to decreasing level in the drain tank. The low level annunciator on CRP 9-7 will actuate. Reduced drainage will be seen in the HP heater(s) due to diversion of this source to the main condenser.

If both the normal and emergency level control valves were to close, the affected moisture separator drain tank level will begin to increase. The affected moisture separator drain tank high level annunciator will actuate.

The emergency drain valve open annunciator will not actuate. The moisture separator high level annunciator on CRP 9-7 will actuate. If level continues to increase, moisture separator high trip annunciator will actuate and a turbine trip, reactor scram will occur.

Removal of the malfunction will restore the affected emergency level control valve back to normal operation.

<u>REFERENCES</u>: P+ID G-191158
CWD B-191301, Sh. 57, 78, 130

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Steam Line "A" Rupture in Drywell

NUMBER: MS06

TYPE: Discrete, Variable (100%=Complete Pipe Degradation, 18 Inch Pipe)

CAUSE: Failure Between RPV and Flow Restrictor

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause main steam line "A" piping to fail to the instructor specified valve. The effects of this malfunction will be dependent upon the severity selected.

At 100% severity, this malfunction will result in a complete severance of main steam line "A" with steam being transferred to the drywell from both ends of the rupture. The MSIV closure (on low pressure) will stop all steam flow except the continuing blowdown through "A" steam line. The reactor will scram from drywell high pressure. Containment isolation groups I through IV will actuate. The large amount of steam lost will cause a rapid depressurization of the reactor vessel. Reactor water level will fluctuate due to changes in mass flow rates and also due to shrink and swell. High drywell pressure will auto initiate the ECCS systems and auto start the standby diesel generators. Suppression pool level and temp. will respond to RPV blowdown, as well as ECCS refilling the vessel. The main generator will trip on MCA trip.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191167
FSAR, Section 14.6.5

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Steam Line "B" Rupture in Steam Tunnel

NUMBER: MS07

TYPE: Discrete, Variable (100%=Double Ended Rupture of 18 Inch Pipe)

CAUSE: Pipe Failure in Steam Tunnel

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause main steam line "B" piping to fail to the instructor specified value. Depending on the severity selected will determine how Group I isolation is actuated. A Group I isolation will occur due to high steam flow, main steam tunnel temperature high or steam line low pressure (if in run mode). The reactor will scram on MSIV closure (less than 90% open), steam flow through the rupture will stop. The gross amount of coolant (steam) lost will have caused rapid depressurization of the reactor until the MSIVs have fully closed. Reactor water level will be unstable due to changes in mass flow rates.

The ECCS systems will auto initiate as required to restore reactor vessel level. Reactor pressure will be controlled by the relief valves. The generator will trip on reverse power, resulting in a turbine trip.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID 191167
OP 3124
FSAR, Section 14.6.5

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Steam Line "C" Rupture in Turbine Building

NUMBER: MS08

TYPE: Discrete, Variable (100%=Double Ended Rupture of 18 Inch Pipe)

CAUSE: Pipe Failure in High Pressure Heater Bay Upstream of Turbine Stop Valves

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause main steam line "C" piping to fail to the instructor specified value. At 100% severity, the rupture will allow steam to flow into the turbine building from the "C" main steam line directly. The high flow in main steam line "C" or Hi area temperature will initiate a Group I isolation causing the MSIVs to close in 3-5 seconds. The reactor will scram as the result of MSIV closure. Reactor water level will then shrink from void collapse and due to the loss of mass from the rupture. Emergency core cooling systems will initiate as required to restore reactor vessel level.

The steam released in the turbine building will be seen as increased turbine building area temperatures, increased flow into the turbine building equipment drain sump and increased radiation levels.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191156, 191167
OP 3124
FSAR 14.6.5

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Gland Seal Regulator Fails Closed

NUMBER: MS09

TYPE: Discrete

CAUSE: Mechanical Failure of the Steam Seal Regulator (CV-1-1A)

PLANT CONDITIONS:

Plant Startup or Turbine @<25% Rated Load.

EFFECTS: This malfunction will result in the closure of the gland seal regulator. Sealing steam pressure to the main turbine on CRP 9-7 will decrease to zero. Steam seal header pressure low annunciator will actuate at 1.7 PSIG on CRP 9-7.

The loss of sealing steam will result in increased air flow to the condenser. Offgas flow rate will increase but will not be able to offset the increased leakage. Condenser vacuum will begin to decrease. If sealing steam is not restored, this condition will result in a turbine trip.

The operator may bypass the failed regulator and restore sealing steam as necessary.

Removal of the malfunction will restore the failed regulator to normal operation.

REFERENCES: P+ID G-191156

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Moisture Separator Drain Tank (A,B,C,D) Normal Level Control Failure

NUMBER: MS10

TYPE: Generic (A,B,C,D), Variable (0-100% Valve Position)

CAUSE: Failure of Level Control (LC24A-D) Output Signal

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the affected normal level control output signal to fail to the instructor specified valve.

If the valve opens to more than that required for normal level, the moisture drain tank level will begin decreasing. The affected moisture separator drain tank low level annunciator on CRP 9-7 will actuate. When it drains completely, the steam delivered to the low pressure turbine will be reduced due to steam blowing by into the affected HP heater. The affected HP heater pressure and level will reflect this additional steam flow.

If the valve closes to less than required for normal level, the moisture drain tank level will begin increasing. The affected moisture separator drain tank high level and moisture separator emergency drain valve open annunciators on CRP 9-7 will actuate. The emergency level control valve will open and will control level at its setpoint. Reduced drainage will be seen in the HP heater(s) due to diversion of this source to the main condenser.

Removal of the malfunction will restore the affected level control valve back to normal operation.

<u>REFERENCES</u>: P+ID G-191158
CWD B-191301, Sh. 57-58

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Main Steam Line "D" Rupture in Drywell (Downstream of Inboard MSIV)

NUMBER: MS11

TYPE: Discrete, Variable (100%=Double Ended Rupture of 18 Inch Pipe)

CAUSE: Pipe Failure Between Inboard MSIV and Primary Containment

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause main steam line "D" piping to fail to the instructor specified value.

At 100% severity, this malfunction will result in a complete severance of main steam line "D" with steam being transferred to the drywell from both ends of the rupture. Flow through the steam lines will increase immediately. Although limited by the flow restrictor, high steam flow will actuate the Group I isolation. All MSIVs will begin to close. The reactor will scram. Flow through the rupture will stop as the MSIVs close in 3-5 seconds. Reactor water level will decrease rapidly as voids collapse and pressure increases. Containment isolation Groups II through IV will actuate due to high drywell pressure. Auto initiation of ECCS systems and auto start the standby diesel generators occur. The main generator will trip on MCA.

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191167
FSAR, Section 14.6.5

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: MSIV (AO-2-80A-D, AO-2-86A-D) Test Mode Failure

NUMBER: MS12

TYPE: Generic (80A-D = A-D, 86A-D = E-H)

CAUSE: Selected MSIV Solenoid Test Pilot Failure

PLANT CONDITIONS:

As Required for the Surveillance

EFFECTS: This malfunction will cause the selected main steam isolation valve (MSIV) test mode operation to fail due to solenoid test pilot failure to close. The affected MSIV will not respond to the event until the associated test pushbutton is depressed. At which time the associated MSIV will continue to travel until it is fully closed and remain closed regardless of the test pushbutton. The plant will respond to the event as evident steam flow and reactor pressure change and reactivity transient.

Removal of the malfunction will return the affected MSIV to normal.

REFERENCES: CWD B-191301, Sh. 1108-1112

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SRM (A,B,C,D) Failure

NUMBER: NM01

TYPE: Generic A-D, Variable (0 - 100% Amplifier Output)

CAUSE: Electronic Failure in the Log Count Rate Amplifier

PLANT CONDITIONS:

Reactor Startup

EFFECTS: The amplifier output will go to the instructor specified value. The log count rate indications will respond linearly to malfunction severity at the rate of about 14.3% severity per decade. The output exceeds the following approximate values. The associated bistable will trip and approximate protective actions and annunciators will actuate:

MF SEVERITY	SRM	TRIP FUNCTION
> 96.9%	$>5\times10^{5}$	HI-HI
> 85.8%	1×10^{5}	HI
> 42.9%	1×10^{2}	RETRACT PERMIT
> 21.4%	3×10^{0}	DOWNSCALE

Removal of this malfunction will restore the amplifier output to normal.

REFERENCES: OP 2130
GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SRM (A,B,C,D) Failure Inoperative

NUMBER: NM02

TYPE: Generic (A,B,C,D)

CAUSE: Auxiliary Relay (K1) Fails De-energized

PLANT CONDITIONS:

Reactor Startup

EFFECTS: This malfunction will result in a high/inoperative status light and annunciator to actuate. It will have no effect on the selected channels level or rate indication. Providing the selected channel is not bypassed, a rod block will be generated.

Removal of the malfunction will restore selected channel to normal operation.

REFERENCES: GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: IRM (A,B,C,D,E,F) Failure

NUMBER: NM02

TYPE: Generic (A,B,C,D,E,F), Variable (0 - 100% Amplifier Output)

CAUSE: Electronic Failure in the Output Amplifier

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: The amplifier output signal will go to the instructor specified value. The range switch will not affect the output signal. If the output exceeds the following values, the associated bistable will trip and appropriate protective actions and annunciators will actuate:

MF SEVERITY	IRM	TRIP FUNCTION
> 96%	>120/125	HI-HI
> 86.4%	>108/125	HI
< 4%	<5/125	DOWNSCALE

Removal of this malfunction will restore the amplifier output to normal.

REFERENCES: OP 2131
GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: IRM (A,B,C,D,E,F) Failure Inoperative

NUMBER: NM04

TYPE: Generic (A,B,C,D,E,F)

CAUSE: Auxiliary Relay (K1) Fails De-energized

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected channel hi-hi/inoperative status light to illuminate. Provided this channel is not bypassed or the mode switch is in run a rod block will occur. Actual neutron flux indication for the channel will not be affected.

The condition will also produce a neutron monitor trip which will result in a trip of the associated scram channel.

If two or more non-associated IRM channels are full scale or inoperative the reactor will scram with the mode switch out of run. If the mode switch is in run, and an associated APRM is downscale a half scram will occur.

Removal of the malfunction will restore affected channel to normal operation.

REFERENCES: GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: APRM (A,B,C,D,E,F) Failure

NUMBER: NM05

TYPE: Generic (A,B,C,D,E,F), Variable

CAUSE: Electronic Failure in Averaging Amplifier

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: The amplifier output will go to the instructor specified value. If the output exceeds the following approximate values, the associated bistable will trip and appropriate protective actions and annunciators will actuate:

MF SEVERITY	APRM	TRIP FUNCTION
> 96%	>120% (or 0.66W + 54%)	HI-HI
> 86.4%	>108% (or 0.66W + 42%)	HI
> 12%	>15%	HI-HI (NOT IN RUN)
> 9.6%	>12%	HI (NOT IN RUN)
< 2%	<2%	DOWNSCALE

After the output has returned to normal, the light indicating that the appropriate bistable has tripped will remain on until manually reset. The trip function will reset automatically.

Removal of this malfunction will restore the averaging amplifier output to normal.

REFERENCES: OP 2132

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: APRM (A,B,C,D,E,F) Failure Inoperative

NUMBER: NM06

TYPE: Generic (A,B C,D,E,F)

CAUSE: Selected Channel Trip Reference Circuit (2 - 34) Fails to Minimum Value

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS:

This malfunction will have no effect on actual flux level indication. The Hi-Hi/Inoperative status light and annunciator will actuate. A rod withdrawal block will be generated as well as neutron monitor trip along with a trip of the associated scram channel(s).

One APRM in each of the non-associated scram channels may be bypassed to clear rod block and half scram condition.

Removal of the malfunction will restore the selected channel to normal operation.

REFERENCES: GE-ED-1974120

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: IRM/APRM Recorder Pen Failure

NUMBER: NM07

TYPE: Generic (7A-52A,B,C,D,E,F)

CAUSE: Pen Servo Drive Motor Bearing Has Frozen (Pen Binding)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected channel recorder pen to freeze in its present position. Any change in flux level will not be observed. The associated alarm and trip functions will respond to actual flux levels.

The IRM/APRM/RBM channel, when selected, will be prevented from indicating actual flux levels. If the alarm level test button is depressed the pen will remain as is.

If power is lost to recorder the pen will remain as is.

Removal of the malfunction will restore recorder to normal operation and pen will indicate actual flux levels.

REFERENCES: CWD B-191301, sh. 692, 693
GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SRM (A,B,C,D) Stuck Detector

NUMBER: NM08

TYPE: Generic (A,B,C,D)

CAUSE: Detector Mechanically Binds In Shuttle Tube

PLANT CONDITIONS:

Reactor Startup/Shutdown

EFFECTS: Upon [illegible]ation of this malfunction the selected SRM channel detector will become stuck in its present position whether static or in travel. The selected SRM detector will not respond to any insert or withdraw commands initiated by the operator. Selected SRM detector flux indication will indicate actual flux variations occurring for that position in the core. All alarm and trip functions will be operable and will respond to detector output.

Removal of the malfunction will restore selected SRM channel detector to normal operation.

REFERENCES: GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: IRM (A,B,C,D,E,F) Stuck Detector

NUMBER: NM09

TYPE: Generic (A,B,C,D,E,F)

CAUSE: Detector Mechanically Binds in Shuttle Tube

PLANT CONDITIONS:

Reactor Startup/Shutdown

EFFECTS: Upon activation of this malfunction the selected IRM channel detector will become stuck in its present position whether static or in travel. The selected IRM detector will not respond to any insert or withdraw commands initiated by the operator. Selected IRM detector flux indication will indicate actual flux variations occurring for that position in the core. All alarm and trip functions will be operable and will respond to detector output.

Removal of the malfunction will restore selected IRM channel detector to normal operation.

REFERENCES: GEK 32440

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: TIP Stuck Detector (1,2,3)

NUMBER: NM10

TYPE: Generic (1,2,3)

CAUSE: Mechanical Binding in Guide Tube

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause selected TIP to become stuck in its present position while inserted in guide tube. Detector will respond to neutron flux variation in the vicinity of the core in which it is stuck. The detector will not respond to insert or withdraw signals including an auto isolation withdraw signal.

Removal of the malfunction will restore the TIP to normal operation.

REFERENCES: GEK 32447B

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: IRM (A,B,C,D,E,F) Noise

NUMBER: NM12

TYPE: Generic (A,B,C,D,E,F)

CAUSE: Selected IRM Channel Spikes Due to Bad Cable Connection

PLANT CONDITIONS:

Reactor Startup/Shutdown

EFFECTS: This malfunction will cause the selected IRM channel(s) to have spurious indication. The affected channel(s) will have erratic spikes every 1 - 10 secs. The IRM's recorder trace will reflect the magnitude of the spikes. The IRM trip circuitry will respond if alarm/trip setpoints are reached. The affected IRM channel may be bypassed if needed. Actual neutron flux will not be affected.

Removal of the malfunction will restore affected IRM channel to normal operation.

REFERENCES: GEK 32447B

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc Flow Converter Failure (A,B)

NUMBER: NM13

TYPE: Generic (A,B)

CAUSE: Electronic Failure of Selected Converter Unit which De-energizes Relays (K3 and K4)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected flow converter unit to fail resulting in an APRM flow ref. Off normal annunciator actuation and rod block initiation. The selected flow converter "comparator" and "UPSCL/INOP" status light will illuminate.

Removal of the malfunction will restore affected flow converter to normal operation.

REFERENCES: GE-ED-197R120
VY 5920-3881, 3882, 3883, 3884
GEK 32438, RMCS
GEK 19356A, APRM - Flow and Auxiliary Unit

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: LPRM (XXYYA) Fails Upscale

NUMBER: NM2

TYPE: Generic

CAUSE: Selected LPRM has Internally Shorted

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the output of selected LPRM going to maximum. Indication of this condition will be indicated on the rod display by the upscale amber status light and the LPRM upscale annunciator. Upon selection of the associated control rod the affected LPRM will indicate full scale.

The appropriate APRM and RBM will respond to the change in LPRM input.

Removal of the malfunction will restore affected LPRM, APRM and RBM to normal operation.

REFERENCES: GE-ED-197R120

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: LPRM (XXYYA) Fails Downscale

NUMBER: NM3

TYPE: Generic

CAUSE: Selected LPRM Flux Amplifier Output Signal Becomes Less than Preset D.C. Reference Voltage

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will result in the output of the selected LPRM going to minimum value. Indication of this condition will be indicated on the rod display by the downscale white status light and the common LPRM downscale annunciator actuating. Upon selection of the associated control rod, the affected LPRM will indicate downscale.

The appropriate APRM(s) and RBM(s) will respond to the change in the selected LPRM input.

Removal of the malfunction will restore affected LPRM, APRM, and RBM to normal operation.

REFERENCES: GE-ED-197R120

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Stack Isolation Vaive (FCV-11) Fails to Closed Position

NUMBER: OG01

TYPE: Discrete

CAUSE: Failure of FSO-11 Resulting in Closure of FCV-11 On Loss of Air

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will result in the stack isolation valve (FCV-11) fail to the closed position due to failure of FSO-11. FCV-11 valve indication will respond to the event, offgas system outlet (SCFM) flow will decrease to zero and offgas system outlet (PSIG) pressure will increase in the positive direction as indication on PI-OG-2004 and PI-OG-07 respectively offgas system pressure will increase with a buildup in SJAE back pressure and eventually causing loss of main condenser vacuum. Loss of condenser vacuum will automatically shutdown the plant.

Removal of the malfunction will re*tore the stack isolation valve FCV-11 to normal allowing the offgas system to be returned to normal.

REFERENCES: 104-00774, Operator Training Course AOG System
P+ID G-191157, 191162
VY-E-75-001 Flow diagram AOG

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Catalyst (A,B) Contamination

NUMBER: OG02

TYPE: Generic (A,B)

CAUSE: Excessive Moisture Introduction

PLANT CONDITIONS:

AOG System In Operation

EFFECTS: This malfunction will result in the selected recombiner subsystem A(B) catalyst to be contaminated due to excessive moisture introduction temperature will decrease to approximately 240°F over a short period of time. Recombiner train flow rate will be dependent on ejector steam supply pressure. Hydrogen concentration will increase from current value. Recombiner pressure drop will increase greater than 1.5 PSI. The affected recombiner may be removed from service in order to sustain plant operations.

Removal of the malfunction will return the affected recombiner catalyst to normal.

REFERENCES: A719-0200, Offgas Treatment System P+ID
104-00774, Operator Training AOG System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Offgas System (A,B) Explosion

NUMBER: OG03

TYPE: Generic (A,B)

CAUSE: Loss of Dilutant Steam

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will result in the selected offgas system (A,B) explosion due to loss of dilutant steam. This event will cause a hydrogen detonation in the air ejector discharge piping. The operating recombiner offgas system will reflect the explosion with a spike in system inlet pressure, preheater inlet pressure, separator outlet and recombiner differential pressure. Interruption in offgas flow will occur for duration of event. The plant may trip due to loss of main condenser vacuum if the offgas system cannot be restored.

Removal of the malfunction will return the selected offgas system to normal.

REFERENCES: 104-00774 Operator Training AOG System
A719-0200, Off Gas Treatment System P+ID

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Heater Drain Tank Pump (A,B) Trip

NUMBER: 0G04

TYPE: Generic (P-151-1A, 1B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the selected offgas recombiner subsystem heater drain tank pump P-151-1A (1B) to trip due to overload. The standby heater drain tank pump will auto start provided control switch on CRP 9-50 is in "Auto" mode. Alarm "high drain tank level" will actuate on increasing level. Preheater efficiency may be affected as evident by possible water-log with resultant lower temperature.

Removal of the malfunction will allow selected heater drain tank pump to operate normal.

REFERENCES: 104-00774, Operator Training AOG System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: AOG Vacuum Pump (A,B) Trip

NUMBER: OG05

TYPE: Generic (VP-1A, 1B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will cause the selected AOG vacuum pump (VP-1A, 1B) to trip if running, due to overload. The vacuum pumps provide the additional boost required to move the offgas through AOG. Automatic switch over of the vacuum pumps is initiated on shutdown of the on-stream pump. Vacuum pump D/P will increase as indicated on DPI-OG-1606A(B). After filter D/P inches WG will change as indicated on DPI-OG-1605A(B) offgas flow to stack via 24 inch delay pipe will decrease as indicated on FR-2201(A) and FI-OG-200 and associated system outlet PSIG will increase as indicated on PI-OG-1307. Annunciation for vacuum pump system will respond to the event. Once the standby vacuum pump is running the affected parameters will return to normal.

Removal of the malfunction will allow the selected vacuum pump to be started.

REFERENCES: 104-00774 Operator Training Course AOG
A719-0200, CVI Off Gas Treatment System P+ID

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: AOG Condensate Booster Pump (A,B) Trip

NUMBER: OG06

TYPE: Generic (P-150-1A, 1B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause the selected AOG condensate booster pump (P-150-1A, 1B) to trip if running due to overload. The AOG condensate booster pump(s) pump condensate from the main condenser via the condensate pump discharge at approximately 120°F and circulated it through the recombiner offgas condensers and back to condensate system upstream of the SJAE. The affected pump flow will decrease as indicated on FIC-9201. If the flow is allowed to decrease below 800 GPM, both offgas recombiner trains will automatically shutdown requiring operator restart. "Shutdown, HE-101 cooling flow low" alarm will actuate after alarm "condenser HE-101 cooling flow low" actuates. Loss of the offgas system will cause subsequent loss of condenser vacuum and eventually a plant trip.

Removal of the malfunction will allow restart of the affected AOG condensate booster pump.

REFERENCES: 104-00774, Operator Training Course AOG
A719-0208, Off Gas Treatment P+ID

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recombiner (A,B) Fails to Auto Shift

NUMBER: OG07

TYPE: Generic (A,B)

CAUSE: Logic Failure

PLANT CONDITIONS:

Off Gas System Operating

EFFECTS: This malfunction will result in the selected recombiner (A,B) failure to auto shift when required. There are six (6) abnormal situations which will cause automatic switch over from the operating to the standby train:

- A) Recombiner inlet temperature less than or equal to 250°F
- B) Recombiner temperature less than or equal to 250°F
- C) 'A' train outlet hydrogen concentration greater than or equal to 2% by volume
- D) 'B' train outlet hydrogen concentration greater than or equal to 2% by volume
- E) Opening of condenser coolant bypass valve (C9002 or C9003)
- F) Loss of power to train

This malfunction will not affect manual switchover of recombiner trains. Associated alarms will actuate as conditions develop. Failure of recombine auto shift will eventually cause loss of hydrogen and oxygen recombination which may result in excessive hydrogen concentration and subsequent loss of condenser vacuum and eventually plant trip.

Removal of the malfunction will return auto shift capability to normal.

REFERENCES: 104-00774, Operator Training Course AOG

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Moisture Separator Drain Valve Fails (A,B) Closed

NUMBER: OG08

TYPE: Generic (OG-402A, B)

CAUSE: Solenoid Valve Failure for Selected Moisture Separator Drain Valve

PLANT CONDITIONS:

Off Gas System Operating

EFFECTS: This malfunction will cause the selected moisture separator [MS-101-1A(B)] drain valve [OG-402A(B)] to fail closed due to failure of associated solenoid valve. Moisture separator tank level will increase with resultant alarm "separator MS-101 water level high" actuation.

Removal of the malfunction will restore the affected moisture separator drain valve [OG-402A,(B)] to normal permitting draining of MS-101-1A(B).

REFERENCES: A719-0200, Off Gas Treatment System P+ID

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recombiner Dual Isolation

NUMBER: OG09

TYPE: Discrete

CAUSE: Logic Failure, Contact Failure 39 (Shutdown Low Condenser Water Flow)

PLANT CONDITIONS:

Reactor Operating At Power, 1-100%

EFFECTS: This malfunction will result in the dual isolation of the offgas recombiners due to logic failure. The inlet isolation valves OG-101A+B will inadvertently close. Pressure rise at the air ejector outlet will be evident on PI-OG-1301. The increase in back pressure will eventually lead to a loss of main condenser vacuum with subsequent plant trip. Preheater inlet pressure will decrease, recombiner inlet flow will decrease, preheater steam supply pressure will decrease, separator outlet pressure will decrease, delay pipe air flow will decrease, regen heater inlet flow will decrease, after cooler outlet pressure will decrease regen outlet pressure will decrease, system outlet pressure will decrease and system outlet flow will decrease.

Removal of the malfunction will allow restart of the recombiner train(s).

REFERENCES: A719-0200, Off Gas Treatment System P+ID

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Stack Isolation Valve (FCV-11) Fails to Close

NUMBER: OG10

TYPE: Discrete

CAUSE: Mechanical Binding

PLANT CONDITIONS:

Off Gas System Operating

EFFECTS: This malfunction will result in failure of the stack isolation valve FCV-11 to close when required due to mechanical binding. Operation of the control switch will not close the valve. Plant protection logic from offgas high radiation will not close the valve. Offgas stack activity and flow rate will not be affected.

Removal of the malfunction will return the offgas stack isolation valve FCV-11 to normal allowing closure with the control switch and auto isolation via the plant protection logic.

REFERENCES: P+ID G-191162

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Off Gas Preheat Steam Valve (A,B) Fails Closed

NUMBER: OG11

TYPE: Generic (A,B)

CAUSE: E/P Converter Failure of Selected Valve (OG-102A,B)

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause the selected off gas preheater steam valve to close. This results in a loss of steam supply to the affected preheater. Recombiner inlet temperature will begin to decrease. Recombiner temperature low annunciator on PNL AL-A will actuate. This results in a loss of recombiner efficiency and subsequent increase in hydrogen concentrations. Due to the decrease in the recombination process, the offgas flow rate will increase. Radiation levels will not change. The operator can place the standby offgas string in service and return the offgas system to normal operations. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore the selected I/P converter to normal and allow the affected offgas string to be returned to normal operation.

REFERENCES: P+ID A719-0200

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Standby Gas Treatment Fan (A,B) Trip

NUMBER: PC01

TYPE: Generic (2A, 2B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Standby Gas Treatment System Operating

EFFECTS: This malfunction w[illegible] cause the selected standby gas treatment fan (2A, 2B) to trip if running. Following fan trip, the 9KW heater will de-energize with subsequent energization of the 1KW heater. Air flow indication will decrease to 0. Associated SBGT fan inlet/outlet isolation valve will close.

Removal of the malfunction will allow restart of the affected standby gas treatment fan.

REFERENCES: 104-0000, Standby Gas Treatment
104-0060, Standby Gas Treatment Operating Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Reactor Building HVAC Supply Fan (A,B)

NUMBER: PC02

TYPE: Generic (A,B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the loss of the selected reactor building HVAC (Heating Ventilation Air Conditioning) supply fan (RSF-1A,1B). Upon failure of either the running exhaust or supply fan the standby fan will be automatically started.

Reactor Building HVAC supply fan RSF-1A and RSF-1B have status indicating lamps on CRP 9-25.

Removal of the malfunction will allow subsequent restart of affected reactor building HVAC supply fan (1A,1B).

REFERENCES: 104-00786, HVAC

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Drywell Cooling Unit (RRU - 1,2,3,4)

NUMBER: PC03

TYPE: Generic (RRU - 1,2,3,4)

CAUSE: Selected RRU 1 - 4 Drive Belt Breakage

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected drywell cooling unit(s) drive belt to break. The air flow through the cooling unit will stop. The differential temperature across the drywell cooling unit will decrease as the outlet duct temperature equalizes with drywell air temperature. Drywell temperature and pressure will increase.

The effects of this malfunction are similar to those of malfunction SW02 effects. Associated annunciators will actuate when appropriate.

Removal of the malfunction will require re-initialization of Simulator.

REFERENCES: P+ID G-191238

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Drywell/Torus Differential Pressure Controller Failure

NUMBER: PC04

TYPE: Discrete, Variable (0 - 100% of Control Range)

CAUSE: Failure of Controller Output

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause the drywell/torus differential pressure controller 1-156-3 output to fail at the selected severity (0 - 100% range of control). Drywell/torus differential pressure is manually maintained at greater than or equal to 1.7 psid in order to reduce damage to suppression chamber internals in the event of a blowdown. The differential pressure minimizes the water level in the downcomer piping and subsequently reduces the water slug upon pressurization. Drywell pressure controller 1-156-3 is used to make-up nitrogen to drywell/torus dependent on setpoint demanded (0 - 100% of scale). The malfunction will remove control from 1-156-3 output. This will effect primary containment parameters such as drywell pressure, torus pressure, torus water level, drywell/torus delta-p, and to a lesser degree oxygen concentration. If failed high, the associated parameters will increase in magnitude.

Removal of the malfunction will return drywell/torus delta-p controller to normal.

REFERENCES: 104-00958, Primary Containment and CAD
P+ID G-191175
CWD B-191302, sh. 750-756

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Safety/Relief Valve (RV2-71A,B,C,D) Line Break

NUMBER: PC05

TYPE: Generic (A,B,C,D)

CAUSE: Piping Failure of Selected Safety/Relief Valve (A,B,C,D) Located in the Torus Above the Water Level

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause the selected safety/relief valve(s) discharge line to break at the location above the torus water level. If the malfunction is active prior to safety/relief valve operation, there will be no observable effects. Upon safety/relief actuation, the following major events will occur:

A. Suppression chamber pressure will increase.
B. Suppression chamber air temperature will increase (also water temperature).
C. Torus/drywell vacuum breakers will operate as required.
D. Torus/drywell delta-pressure will be lost, drywell press & temp. will increase as long as vacuum breakers remain lifted.

This malfunction can be removed by re-initialization of the Simulator.

REFERENCES: FSAR 14.6.5
GEK 32437, Automatic Blowdown System

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Primary Containment Rupture

NUMBER: PC06

TYPE: Discrete, Variable (100% = 18 inch pipe diameter)

CAUSE: Seismic Event Causes SB-16-19-7A to Open/Associated Downstream Flange Fails

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the primary containment to rupture (for 100% severity) due to a seismic event causing (SBGT) SB-1C-19-7A valve to open and its associated downstream valve flange to fail. Drywell pressure will decrease and drywell-torus delta-pressure will reflect the event. The SB-16-19-7A valve position will indicate open. Primary containment integrity can not be maintained unless the affected line is isolated.

Removal of the malfunction will return the affected valve/flange to normal, once the valve is positioned to close.

REFERENCES: CWD B-191301, sh. 1113

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Secondary Containment Rupture

NUMBER: PC07

TYPE: Discrete

CAUSE: Reactor Building Blowout Panel Failure

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause the secondary containment to rupture due to reactor building blowout panel failure. Secondary containment integrity can not be maintained. Applicable control room instrumentation will reflect the event. Standby gas treatment system will not be able to pull normal vacuum on the Rx building.

Removal of the malfunction will return the Rx building blowout panels to normal.

REFERENCES: Lot-03-206, Secondary Containment

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW LEAKAGE INTO THE DWFDS

NUMBER: PC08

TYPE: VARIABLE 100% = 7.2 gpm

CAUSE: Drywell RRU's cooling coils leaking RBCCW into DWFDS.

PLANT CONDITIONS:

All

EFFECTS: If this malfunction is entered at a high enough severity it will cause the DWFDS leakage alarms to actuate. An increase in pumpdowns will occur and the DWFDS totalizer will increase. This malfunction will affect the RBCCW surge tank level and along with other leakage may cause the surge tank lo level alarm to annunciate.

REFERENCES: FSAR Section 10

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW LEAKAGE INTO THE DWEDS

NUMBER: PC09

TYPE: VARIABLE 100% = 28.8 gpm

CAUSE: DWEDS cooling coil leaking.

PLANT CONDITIONS:

All

EFFECTS: If entered at a high enough severity this malfunction will cause the actuation of the DWEDS leak rate alarms and increase the pumpdown rate. The DWEDS totalizer will increase by the amount of leakage. This malfunction will affect RBCCW surge tank level and along with other RBCCW leakage may cause the low level alarm to actuate.

REFERENCES: FSAR Section 10

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: ISOLATION VALVE (AABBC) FAILURE TO CLOSE

NUMBER: PC1

TYPE: GENERIC

CAUSE: FAILURE OF 42 "O" RELAY IN CLSOE CKT OF MOV'S
FAILURE OF AIR TO VENT OFF OF AOV'S

PLANT CONDITIONS:

ANY PLANT CONDITIONS

EFFECTS: THE FOLLOWING ISOLATION VALVES WILL FAIL TO CLOSE BY EITHER MANUAL OR AUTOMATIC MEANS

CU15 CU18 RC15 RC16 HP15 HP16 RH17 RH18 LW82 LW83 LW94 LW95 SB06 SB06A SB06B SB07 SB07A SB07B SB09 SB10 SB11 SB12 SB20 SB22B

REMOVAL OF THE MALFUNCTION WILL ALLOW CLOSURE OF THE VALVE.

REFERENCES:

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Turbine Trip

NUMBER: RC01

TYPE: Discrete

CAUSE: Short Circuit of Push Button S-17 (turb trip push button) on CRP9-4

PLANT CONDITIONS:

RCIC System Operating

EFFECTS: This malfunction will cause the RCIC turbine to trip. The RCIC turbine trip and throttle valve will close. Turbine speed will coast down pump flow and discharge pressure will decrease. RCIC turbine trip annunciators will actuate. Reactor vessel level will reflect the loss of this supply water if injecting to vessel.

Removal of the malfunction will restore the RCIC turbine to normal operation.

REFERENCES: GEK-9614
CWD B-191301, sh. 1181

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Failure to Auto Start

NUMBER: RC02

TYPE: Discrete

CAUSE: 13A-K1, K2 Relay Failure Inhibits RCIC Auto-Start

PLANT CONDITIONS:

Reactor Vessel Low Level

EFFECTS: This malfunction will result in the failure of the RCIC system to auto start on an initiation signal. Reactor parameters will reflect the loss of this system and will be dependent on remaining equipment in operation. The operator will have the capability of manually starting the RCIC system and inject into the vessel.

Removal of the malfunction will return relays 13A-K1, K2 to normal operation.

REFERENCES: GEK-9614
CWD B-191301, sh. 1180

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Flow Controller Failure

NUMBER: RC03

TYPE: Variable (0 - 100% of Range)

CAUSE: Controller Output Signal Failure in Auto Mou.

PLANT CONDITIONS:

RCIC System Operating

EFFECTS: This malfunction will cause the RCIC flow controller output signal too fail to the severity selected. If the RCIC flow control signal fails low, the RCIC turbine speed will decrease to its low speed limit if operating in auto. System flow and pressure will reflect the turbine speed decrease. If the malfunction is activated prior to a system start, the RCIC turbine will increase to its low speed stop and remain at that speed.

If the RCIC flow control signal fails high, the RCIC turbine speed will increase to its high speed limit if operating in auto, system flow and pressure will reflect the turbine speed increase. If this malfunction is activated before the system starts, the turbine will increase in speed and may trip due overspeed.

The operator may place the controller in manual and adjust the flow controller to the desired value.

Removal of the malfunction will restore the controller output to normal.

REFERENCES: P+ID G-191174
GEK-9614
CWD B-191301, sh. 1184

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Inadvertent Initiation

NUMBER: RC04

TYPE: Discrete

CAUSE: Electrical Short In Vessel Low Level Sensing Circuit Resulting In Energizing Relays 13A-K1, K2

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the RCIC system to automatically start, turbine speed and pump discharge pressure will begin to increase. When RCIC pump discharge pressure overcomes reactor pressure the system will begin to inject CST water into the vessel. This will result in a reactivity addition. The feedwater control system will compensate for the increase in reactor water level by decreasing the input from the feedwater system. Reactor water level will return to approximately the original value.

Removal of the malfunction will restore the failed relays to normal.

REFERENCES: GEK-9614
CWD B-191301, sh. 1180

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Inadvertent Isolation

NUMBER: RC05

TYPE: Discrete

CAUSE: Failure of DPIS-13-84

PLANT CONDITIONS:

Reactor Startup of Power Operation

EFFECTS: This malfunction will result in the loss of steam supply to the RCIC turbine due to closure of the inboard and outboard steam isolation valves. The RCIC steam line hi diff pressure annunciator will actuate. RCIC turbine speed will decrease as a function of the decreasing steam supply pressure and turbine coast down characteristics. RCIC pump discharge pressure and system flow will decrease correspondingly. The RCIC turbine will trip as a result of the auto isolation.

Removal of the malfunction will restore the RCIC system to normal operation.

REFERENCES: GEK-9614
CWD B-191301, sh. 1179

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Exhaust Diaphragm Failure

NUMBER: RC06

TYPE: Discrete

CAUSE: Exhaust Diaphragm Ruptures

PLANT CONDITIONS:

RCIC System Operating

EFFECTS: This malfunction will cause the RCIC exhaust diaphragm to rupture resulting in steam being released to the torus area atmosphere. Radiation monitoring system will detect an increase in activity. RCIC flow controller will maintain system flow constant in spite of increased RCIC turbine steam flow. When the steam leak detection system high temperature setpoint is reached, the RCIC system will automatically isolate. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. Turbine exhaust pressure indicated on CRP 9-4 will decrease. All associated alarms will actuate when appropriate.

Removal of the malfunction will restore the RCIC system to normal operation.

REFERENCES: GEK-9614
P+ID G-191174

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Injection Valve (RCIC-21) Fails to Auto Open

NUMBER: RC07

TYPE: Discrete

CAUSE: RCIC-21 Auto Open Contacts (6-5) Fail Open

PLANT CONDITIONS:

RCIC System Auto Initiation

EFFECTS: This malfunction will allow the RCIC system equipment to operate as normal on an initiation signal with the exception of valve RCIC-21 remaining closed. This will prevent water from being admitted to the reactor. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. The operator can manually open RCIC-21 from CRP 9-4 and restore flow to vessel.

Removal of the malfunction will restore RCIC injection valve to respond to auto initiation signal.

REFERENCES: P+ID G-191174, sh. 2
GEK-9614
CWD B-'9130, sh. 1192

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Speed Control Fails (High/Low)

NUMBER: RC08

TYPE: Discrete; A = High; B = Low

CAUSE: Control Signal From Speed Governor Fails to Maximum/Minimum Output in "Auto" Mode

PLANT CONDITIONS:

RCIC System Operating

EFFECTS: This malfunction will cause the RCIC speed control to fail:

A. RCIC speed control fails high: This malfunction will cause the RCIC turbine speed to increase to its high speed limit if operating in "auto". System flow and pressure will reflect the turbine speed increase. If this malfunction is activated before the system starts the turbine will increase in speed and may trip due to overspeed. Manual operation of turbine speed is not affected.

B. RCIC speed control fails low: This malfunction will cause the RCIC turbine speed to decrease to its low speed limit if operating in "auto". System flow and pressure will reflect the turbine speed decrease. If the malfunction is activated prior to a system start, the RCIC turbine will increase to its low speed stop and remain at that speed. Manual operation of turbine speed is not affected.

Removal of the malfunction will restore the controller output to normal.

REFERENCES: GEK-9614, RCIC
CWD B-191301, sh. 1179, 1184

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RCIC Steam Line Leak

NUMBER: RC09

TYPE: Discrete, Variable (100% = 2 inch Pipe)

CAUSE: Piping Failure on RCIC Steam Supply Line in the RCIC Room between the RCIC-131 and RCIC-1 Valves

PLANT CONDITIONS:

RCIC System Operating

EFFECTS: This malfunction will cause the RCIC room temperatures to increase above conditions for RCIC isolation. Upon isolation, if the RCIC turbine is operating, the RCIC turbine will trip and system flow/discharge pressure will drop off. If the RCIC turbine is not operating, the ambient temperature will slowly return to normal. All appropriate annunciators will actuate on high area temperature and isolation.

Removal of the malfunction will allow the RCIC system returned to normal operation.

REFERENCES: GEK-9614, RCIC
CWD B-191301, sh. 1179
P+ID G-191174

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: CRD Pump (A,B) Trip

NUMBER: RD01

TYPE: Generic (A,B)

CAUSE: Activation of Overcurrent Device (50)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: If the pump is running, or when it is started, the breaker will trip due to overcurrent. If the alternate CRD pump is not started, a total loss of CRD system flow and pressure occurs appropriate annunciation indicating the selected CRD pump has tripped. CRD system flow, cooling water flow and drive water flow will decrease to zero. Pressure throughout the system will decrease to static pressure. Charging water pressure will decrease to pump suction pressure. Accumulator pressure will begin to decay resulting in random accumulator lamps actuating. Loss of cooling to the CRD drives will result in CRD high temperature annunciator actuating.

Removal of the malfunction will allow the affected CRD pump to be restarted.

REFERENCES: CWD B-191301, sh. 884-885
OP 2111
P+ID G-191170

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Stuck

NUMBER: RD02

TYPE: Generic

CAUSE: Mechanical Binding

PLANT CONDITIONS:

Reactor Startup or Operating at Power

EFFECTS: The selected control rod will stick at its present position. CRD drive flow will indicate stall conditions if any normal rod motion is requested. If the rod is scrammed, it will not move but will pass seal leakage flow to the scram discharge volume allowing the HCU accumulator to discharge to reactor pressure and then allow flow from the reactor vessel.

Removal of the malfunction will allow the control rod to move normally.

REFERENCES: P+ID G-191170
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Uncoupled

NUMBER: RD03

TYPE: Generic

CAUSE: Coupling Spud Failure

PLANT CONDITIONS:

Reactor Startup and Power Ascension

EFFECTS: During normal rod withdrawal this malfunction will allow the control rod drive mechanism to withdraw beyond notch 48 and settle into the overtravel out position, actuating the overtravel annunciator.

If this malfunction is activated simultaneously with the stuck rod malfunction RD02, the drive mechanism can be separated from the control rod blade. Subsequent removal of the malfunction RD02 will allow the control rod to drop to the current drive mechanism position.

Removal of the malfunction will restore normal operation of the coupling spud. The control rod must be inserted at least one notch to recouple the rod.

REFERENCES: GEK 32424A - 780

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Drift In

NUMBER: RD04

TYPE: Generic, Variable (100% = Half Normal Drive Speed)

CAUSE: Leaking Scram Outlet Valve

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the rod display drift lamp and the common rod drift annunciator to actuate. The four rod display will indicate the selected rod drifting in at the severity selected by the instructor.

The control rod will respond to normal in or out motion when actuated by the operator. However, the control rod will continue to drift in upon completion of the rod settle sequence. Any rod blocks actuated during this malfunction will have no effect on the drifting rod. The control rod can drift to the fully inserted position and if fully inserted, will go to the overtravel in position.

Reactor power level in the local area of the control rod and on a whole core basis will respond to reactivity changes resulting from drifting rod movement.

Removal of this malfunction will allow the control rod to settle at the next available "Even" notch.

REFERENCES: OP 2111

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Drift Out

NUMBER: RD05

TYPE: Generic, Variable (100% = Half Normal Drive Speed)

CAUSE: Drive Mechanism Collet Piston Stuck in the Up Position

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the rod display drift lamp and the common rod drift annunciator to actuate the four-rod display will indicate the selected rod drifting out at the severity selected by the instructor.

The control rod will respond to normal in or out motion when actuated by the operator, however, the control rod will continue to drift out upon completion of the rod settle sequence. Any rod blocks actuated during this malfunction will have no effect on the drifting rod. The control rod can drift to the fully withdrawn position.

Reactor power level in the local area of the control rod and on the whole core basis will respond to reactivity changes resulting from drifting rod movement.

Removal of the malfunction will allow the control rod to settle at the next available "Even" notch.

REFERENCES: OP 2111
GEK 32438, 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Scram

NUMBER: RD06

TYPE: Generic (XX-YY)

CAUSE: Blown Fuse

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected control rod to scram. The rod display scram, drift, accumulator and full-in indications will actuate along with the associated annunciators. If the selected control rod has been selected the 4-rod display will indicate fully inserted position. Neutron flux will indicate this condition locally and on a whole core basis dependent upon rod worth. The scram discharge volume should not fill to the alarm point since the drain valves do not close.

Removal of the malfunction will permit normal rod movement from its fully inserted position.

REFERENCES: P+ID G-191170
CWD B-191301, sh. 817

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Accumulator Low Pressure

NUMBER: RD07

TYPE: Generic (XX-YY)

CAUSE: Nitrogen Leak

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: The nitrogen pressure in the specified control rod drive accumulator will immediately drop to atmospheric pressure, causing the low pressure annunciator to actuate. No effects will be observed on normal rod motion.

If the rod is scrammed, its only driving force will be reactor pressure. This will cause a slower than normal rod insertion time. Below about 450 psig reactor pressure, the rod will not move at all.

Removal of the malfunction will restore the accumulator nitrogen pressure to normal.

REFERENCES: GEK 32424A, 780, 9582
P+ID G-191170
OP 2111

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Position Indication Failure at Next Even Position

NUMBER: RD08

TYPE: Generic (XX-YY)

CAUSE: The Next Reed Switch That Should Close Will Fail Open

PLANT CONDITIONS:

Reactor Startup/Shutdown

EFFECTS: No effects will be seen until the specified rod is moved. When the rod is moved and settles into its next even position, no rod position will be indicated on the full core display or on the four-rod matrix. When the settle timer times out, a rod drift alarm will occur since no even reed switches are closed and the rod is not selected and driving. Rod block annunciator is also actuated.

NOTE: This malfunction will affect either the next higher or next lower even numbered reed switch for the specified rod. The direction of motion will determine which one will be affected. Once one of them is affected, the other will continue to operate normally.

Removal of the malfunction will allow the reed switch to operate normally.

REFERENCES: GEK 32407

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Scram Discharge Volume Drain Valve (33A,33B,33C,33D) Fails Open

NUMBER: RD09

TYPE: Generic (33A,33B,33C,33D)

CAUSE: Selected AOV Drain Valve Stem Binds, In Open Position

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause selected SDV drain valve not to close regardless of plant condition. No effects will be seen until the reactor scrams. Those HCU's discharging to the selected SDV will drain to the reactor building equipment drain sump. As the water flashes to steam, temperature and activity in the sump area will increase. Water leaking past seals in the drives will continue to flow into the SDV until scram signal is reset. The red (Full-open) valve position indicating lamp(s) will remain illuminated. Scram discharge volume isolation test will have no effect on stuck valve.

Removal of the malfunction will restore SDV drain valve to normal operation.

REFERENCES: P+ID G-191170
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Scram Discharge Volume Drain Valve (33A,33B,33C,33D) Fails Closed

NUMBER: RD10

TYPE: Generic (33A,33B,33C,33D)

CAUSE: Selected AOV Drain Valve Diaphragm Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: Any water entering the scram discharge header will not drain from the instrument volume. Water may enter the scram discharge header as the result of scrammed control rods or from scram valve leakage (RD04). As level increases, the scram discharge volume not drained annunciator will actuate at approximately 12 gallons a rod block and rod block annunciator actuate. When level increases to approximately 21 gallons, a reactor scram is initiated and the scram discharge volume Hi scram annunciator actuates.

When the scram is reset the discharge volume will not drain.

Removal of the malfunction will restore selected AOV diaphragm to normal.

REFERENCES: GEK 32424A
P+ID G-191170
OP 2111

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: CRD Flow Control Valve (A,B) Fails Closed

NUMBER: RD11

TYPE: Generic (A,B)

CAUSE: E/P Converter Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected CRD flow control valve (19A,B) to close. This will result in a decrease in CRD system flow and an increase in charging water header pressure. The system flow will decrease to 5 gpm and drive water header and cooling water header differential pressure will decrease to a very low value.

Movement of control rods by normal operation is impossible since no differential pressures can be developed across the piston in the CRD mechanism. The ability to scram the rods will not be affected. Due to a lack of cooling water the rod drive mechanism temperatures will increase and actuate the high temperature annunciator. Use of flow control valve switch via IDA will permit the unaffected valve to be selected.

Removal of the ma function will return the affected CRD flow control valve to normal.

REFERENCES: P+ID G-191170
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Partial Scram (A,B)

NUMBER: RD12

TYPE: Generic (A,B)

CAUSE: Blockage of Scram Discharge Volume (A,B)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will prevent the control rods from being fully inserted on a scram signal due to reduced capacity in SDV due to blockage. All indications will verify receipt of scram signal (Auto, Manual, Mode Switch to shutdown), de-energizing of scram solenoids, energization of backup scram valves, opening of scram inlet and outlet valves for each CRD/HCU. The SDV instrument volume will not be affected by malfunction. The full core display will verify that all control rods have not been fully inserted to 00 position. Control rods will travel up to 120 inches from their initial positions and will reflect this new position, for the associated SDV. Plant parameters will respond dynamically to the event that generated the scram signal. Individual systems response will depend upon current plant conditions.

Removal of the malfunction will remove blockage in SDV.

REFERENCES: P+ID G-191170

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

<u>TITLE</u>: Scram Discharge Volume (A,B) Leak

<u>NUMBER</u>: RD13

<u>TYPE</u>: Generic, Variable (100% = 1 inch pipe)

<u>CAUSE</u>: Failure of the Weld on the Scram Discharge Instrument Volume Pipe.

<u>PLANT CONDITIONS</u>:

Reactor Startup or Power Operation

<u>EFFECTS</u>: This malfunction will cause the a[illegible]ed scram discharge volume drain line to leak [illegible] a severity selected by the instructor. No effects will be seen until the reactor scrams. At the severity selected, the water will drain to the reactor building floor drain sump. As the water flashes to steam, temperature and activity in the area of the discharge volume will increase. Water will continue to flow to the reactor building due to drive seal leakage until the scram signal is reset.

The Simulator must be reinitialized to remove this malfunction.

<u>REFERENCES</u>: P+ID 191170

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Control Rod (XX-YY) Position Indication Failure at Present Position

NUMBER: RD14

TYPE: Generic (XX-YY)

CAUSE: The Reed Switch at the Present Position Fails Open

PLANT CONDITIONS:

Reactor Startup/Shutdown

EFFECTS: This malfunction will result in the selected control rod (XX-YY) position indication to fail at present position caused by the associated reed switch failing open. This malfunction is similar to malf RD08 except that the reed switch fails at its present position.

Removal of this malfunction will allow the reed switch to operate normally.

REFERENCES: GEK 32407

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: CRD Flow Controller Failure

NUMBER: RD15

TYPE: Variable (0 - 100% Valve Position)

CAUSE: Failure of Controller Automatic Output Signal

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the CRD flow controller auto output signal to fail to the severity selected by the instructor.

If the controller output is lower than normal, the flow control valve will close down. Reduced pressures and flow will occur. Rod drive temperatures will increase and may actuate the high temperature annunciator. Attempting to move rods with low drive pressure will result in lower than normal rod speeds or no movement at all. If the controller output signal causes the flow control valve to close, movement of control rods by normal operation will not be possible.

If the controller output is higher than normal, the flow control valve will open. Increased pressure and flows will occur. Drive temperatures will decrease. High drive pressure will result in higher than normal rod speeds, possibly double notching in.

Placing the controller in manual will allow the operator to restore normal flow.

Removal of the malfunction will restore normal operation of the flow controller automatic output.

REFERENCES: P+ID G-191270
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: CRD Stabilizing Valve (1A,2A,1B,2B) Fails to Close

NUMBER: RD16

TYPE: Generic (1A,2A,1B,2B)

CAUSE: Selected Solenoid Operated Valve A: 25A-8A/B, B: 25B-8A/B Sticks

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause the selected stabilizing valve not to close once an insert/withdrawn signal has been initiated. Depending on which stabilizing valve is selected (insert/withdraw) system flow will increase by 4/2 gpm and the flow control valve will close to restore flow to setpoint. Cooling flow will decrease by 4/2 gpm. Drive pressure will decrease slightly and charging pressure will increase slightly.

The operator may select the other set of stabilizing valves to restore normal operation of the system.

Removal of the malfunction will restore normal operation of the solenoid operated valve.

REFERENCES: P+ID G-191170
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Total Failure of Manual Rod Control

NUMBER: RD17

TYPE: Discrete

CAUSE: Power Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in the failure of the reactor manual control system due to a power failure. This failure will result in a loss of control rod positioning capability. The rod power lamp will extinguish and the selected rod will be de-selected on the full core display and 4-rod display. Any selected control rod will not respond to any RMCS signal. It will not be possible to insert control rods if required with the Emergency In/Notch Override Switch. When control rod insertion is required to assure safe reactor operations, the reactor will have to be scrammed manually.

Removal of the malfunction will restore RMC system power to normal.

REFERENCES: GEK 32438

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Scram Discharge Volume Vent Valve (A,B) Fails Open

NUMBER: RD18

TYPE: Generic (A,B)

CAUSE: Selected AOV Vent Valve Stem Binds in the Open Position

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will cause selected SDV vent valve not to close regardless of plant conditions. No effects will be seen until the reactor scrams. Those HCUs discharging to the selected SDV will pass water to the reactor building exhaust fan suction plenum. Activity in the reactor building exhaust ventilation will increase. Water leaking past the seals in the drives will continue to flow into the SDV until scram signal is reset. The red (Full-open) valve position indicating lamp(s) will remain illuminated. Scram discharge volume isolation test will have no effect on stuck valve.

Removal of the malfunction will restore SDV vent valve to normal operation.

REFERENCES: P+ID G-191170
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RMCS Timer Malfunction

NUMBER: RD19

TYPE: Discrete

CAUSE: Master Timing Circuit Fails to De-energize Withdraw Bus Before 2.0 Second Auxiliary Timer Drops Out

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in a timer malfunction rod select block. The control rod will settle into the next even notch. The rod will not respond to any insert or withdraw commands. The selection of a different rod will not be permitted.

Removal of the malfunction will allow the rod select block to be reset.

REFERENCES: GEK 32438
GEK 32424A

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Pump (A,B,C,D) Trip

NUMBER: RH01

TYPE: Generic (A,B,C,D)

CAUSE: Actuation of Instantaneous Overcurrent Device (50)

PLANT CONDITIONS:

RHR Pump Running

EFFECTS: This malfunction will cause the selected RHR pump, if running or when started, to trip. The pump tripped annunciator will actuate. RHR system flow and pressure will reflect this loss. If the affected RHR pump was being used to reflood the vessel, water level will increase at a lower rate or stop increasing depending on status of other ECCS equipment. If the affected pump was being used for shutdown cooling, the reactor coolant temperature will decrease at a slower rate or start increasing depending on the amount of decay heat and status of other plant equipment.

Removal of the malfunction will allow the affected RHR pump to be restarted. If an auto start signal is present the pump will auto start after the trip is cleared by taking the control switch to stop or pull to lock.

REFERENCES: P+ID G-191172
CWD B-191301, sh. 1300-1303

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Heat Exchanger (A,B) Tube Leak

NUMBER: RH02

TYPE: Generic (A,B), Variable (100% = 20 gpm at Normal Differential Pressure)

CAUSE: Tube Gasket Failure

PLANT CONDITIONS:

Shutdown Cooling, Torus Cooling, Cont. Spray

EFFECTS: This malfunction will cause the selected RHR heat exchanger gasket to leak at the severity selected. Normally leakage will occur from the RHR service water into the RHR system. The affected RHR system conductivity will increase to alarm trip setpoint and annunciate on CRP 9-3. Reactor water level will begin too increase if S/D cooling is in service. Torus level will slowly rise if torus cooling or cont. spray are in service. The affected RHR HX shell side to tube side differential pressure controller and indication will reflect this leakage and try to maintain the proper delta P, if in automatic. If leakage is reversed and is now occurring from RHR system to RHR service water system, an increase in the service water radiation monitoring system will be reflected. Isolation of the affected RHR heat exchanger via IDA will stop the leakage.

Removal of the malfunction will restore the affected RHR HX to normal operation.

REFERENCES: P+ID G-191172
OP 2125

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Containment Spray Valve (26A,26B) Fails to Open

NUMBER: RH03

TYPE: Generic (A,B)

CAUSE: Valve Mechanically Binds in the Closed Position

PLANT CONDITIONS:

Containment Spray Initiation

EFFECTS: This malfunction will cause the selected RHR containment spray valve to mechanically bind in its closed position. When attempting to open the selected valve, via the remote manual switch on CRP 9-3, the valve will remain closed and trip on activation of thermal relay (49). (Trips on thermal due to failure of torque switch). The selected valve position lights will extinguish and the RHR system valve overload annunciator will actuate. This malfunction will only affect valves 26A, B when they are in the full closed position.

Removal of the malfunction will reset device (49) and allow the RHR containment spray valve(s) to operate normally.

REFERENCES: P+ID G-191172
CWD B-191301, sh. 1274 - 1275

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR SW-89 Valve Controller (A,B) Failure

NUMBER: RH04

TYPE: Generic (A,B), Variable (0-100% of Range)

CAUSE: Controller Output Signal Fails in Auto

PLANT CONDITIONS:

Shutdown Cooling

EFFECTS: This malfunction will cause the selected RHR SW-89 Valve Controller output signal to fail (in auto) to severity selected. This will result in the increase/decrease position of SW-89 valve affecting the differential pressure between the tube side and shell side of the RHR heat exchanger(s), normally 20 psid is maintained.

The change in differential pressure will be indicated by the RHR/SW outlet temperature, RHR/SW flow through the RHR HX, and pressure, and the primary coolant temperature. All associated alarms will actuate when appropriate. The operator may control RHR HX tube/shell differential pressure by placing the affected controller in manual and adjusting the DP accordingly.

Removal of the malfunction will return the affected controller auto output signal to normal.

REFERENCES: P+ID G-191172
OP 2124
CWD B-191301, sh. 1286, 1287, 1254, 1259, 1294, 1295

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Shutdown Cooling Isolation Valve (18) Fails Shut

NUMBER: RH05

TYPE: Discrete

CAUSE: RHR System Isolation Valves Control Relay 16A-K29 Contacts (4-3) Fail Close

PLANT CONDITIONS:

Shutdown Cooling

EFFECTS: This malfunction will cause the RHR/SDC isolation valve to fail to the closed position. The RHR pumps running for S/D cooling service will automatically trip due to valve interlocks. RHR 18 valve will indicate closed on CRP 9-3. The flow to the RHR heat exchanger(s) will go to zero resulting in a loss of decay heat removal capability. Reactor vessel temperature will increase dependent upon decay heat generation and remaining equipment in operation. All associated annunciators will actuate when appropriate.

Removal of the malfunction will allow the RHR/SDC isolation valve to be reopened and resume shutdown cooling mode.

REFERENCES: P+ID G-191172
CWD B-191301, sh. 1309, 1314

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Pump (A,B,C,D) Fails to Auto Start

NUMBER: RH06

TYPE: Generic (A,B,C,D)

CAUSE: Auto Start Relay 10A-K12A or 10A-K12B or 10A-K49A or 10A-K49B Contacts (1-2) Fail Open

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the affected RHR pump not starting when an auto initiation signal is present. The automatic functions of the RHR system will occur normally. The operator may start the pump manually and provide coolant to the reactor vessel.

Removal of this malfunction will allow the pump to start automatically if an auto initiation signal is present. If the pump is already running, no effects will be seen from either activation or removal of this malfunction.

REFERENCES: P+ID G-191172
GEK 9611
CWD B-191301, sh. 1250 - 1251, 1255 - 1256, 1300 - 1303

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

__TITLE:__ RHR Injection Valve (27A,27B) Fails to Auto Open

__NUMBER:__ RH07

__TYPE:__ Generic (A,B)

__CAUSE:__ Valve V10-27A,B Auto Open Contacts (4-3) Fail Open

__PLANT CONDITIONS:__

LPCI Auto Injection

__EFFECTS:__ This malfunction will allow the RHR system equipment to operate as normal on an initiation signal with the exception of valve 27A and/or 27B remaining closed. This will prevent water from being admitted to the reactor. Reactor water level will reflect the loss of this water supply and will depend on remaining equipment in operation. The operator can manually open the affected valve from CRP 9-3, pressure permitting, and restore flow to the vessel.

Removal of the malfunction will restore RHR injection valve(s) to respond to auto initiation signal.

__REFERENCES:__ CWD B-191301, sh. 1272 - 1273
GEK 9611
P+ID G-191172

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Process Radiation Monitor Failure

NUMBER: RM01

TYPE: Generic, Variable (0 - 100% of Meter Scale)

CAUSE: Detector Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected process radiation monitor will fail to the instructor specified percent of the range. If a range switch is associated with the selected process radiation monitoring channel, the range switch must be operated to find the on-scale position for the selected channel. Any system or reactor protection functions associated with the selected monitor will alarm or trip at the normal setpoint. Malfunction values will appear on recorders associated with the process radiation monitors.

RM01A - OFF GAS A
RM01B - OFF GAS B
RM01C - FLUX TILT
RM01D - STACK GAS 1
RM01E - STACK GAS 2
RM01F - STACK GAS 3
RM01G - MN STMLN CH A
RM01H - MN STMLN CH B
RM01I - MN STMLN CH C

RM01J - MN STMLN CH D
RM01K - RAD WASTE EFFLUENT
RM01L - SW EFFLUENT
RM01M - CCW EFFLUENT
RM01N - DISCHARGE CANAL
RM01O - RX BL VNT R/F F/R
RM01P - RX BL VNT R/F F/R
RM01Q - RX BL VN RX ZONE A
RM01R - RX BL VN RX ZONE B
RM01S - CONT MONITOR PART
RM01T - CONT MONITOR GAS
RM01U - RX BLG VENT PART
RM01V - RX BLG VENT GAS
RM01W - INLET GARD BED A
RM01X - INLET GARD BED B
RM01Y - INLET ABSORBER A
RM01Z - OUTLET ABSORBER A
RM01AA - OUTLET ABSORBER B
RM01BB - TO HOLD UP AND STACK A
RM01CC - TO HOLD UP AND STACK B
RM01DD - VENT EXH PART
RM01EE - VENT EXH GAS
RM01FF - DRYWELL RANGE CH 1
RM01GG - DRYWELL RANGE CH 2

Removal of the malfunction will restore the selected process radiation monitor channel to normal operation.

REFERENCES: OP 3127
GEK 32436
CWD B-191301, sh. 975-978, 981, 983-985
VYNPC Simulator Malfunction Cause and Effects

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Area Radiation Monitor Failure

NUMBER: RM02

TYPE: Generic, Variable (0 - 100% of Process Range)

CAUSE: Detector Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected area radiation monitor will fail to the instructor specified percent of meter scale. The selected area radiation monitor alarm will actuate at the appropriate high or low alarm point.

RM02A - EL-232 TORUS CATWALK
RM02B - EL-252 BY ELEVATOR
RM02C - EL-252 BY RR AIRLOCK
RM02D - EL-252 BY TIP ROOM
RM02E - EL-252 BY DWL AIRLOCK
RM02F - EL-280 BY ELEVATOR
RM02G - EL-252 CRD REPAIR ROOM
RM02H - EL-303 BY ELEVATOR
RM02I - EL-303 RWCU PUMP ROOM
RM02J - EL-318 BY ELEVATOR
RM02K - EL-318 RWCU PRECOAT
RM02L - EL-345 BY ELEVATOR
RM02M - TURB BLDG EL-228 ON NORTH WALL
RM02N - REFUEL FLR WEST
RM02O - SPENT FUEL POOL

RM02P - BY VENT DUCT EL-318
RM02Q - SAMPLE SINK RADWS
RM02R - DECON AREA RADWS
RM02S - LOWER LEVEL RADWS
RM02T - OVER FW PUMP RM
RM02U - STM LINE STAIRS
RM02V - CONDEMIN PNL
RM02W - MACHINE SHOP
RM02X - #2 CIV RAILING
RM02Y - OUTSIDE CTRL RM
RM02Z - MUD TRAIN AREA
RM02AA - AOG CONTROL AREA
RM02BB - AOG STOR AREA

Removal of the malfunction will restore the selected area radiation monitor to normal operation.

<u>REFERENCES</u>: CWD B-191301, sh. 990-999
OP 2135
GEK 32429

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Reactor Bldg Hi Range ARM Failure

NUMBER: RM03

TYPE: Generic, Variable (0 - 100% of Meter Scale)

CAUSE: Detector Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected area radiation monitor will fail to the instructor specified percent of meter scale. The selected area radiation monitor alarm will actuate at the appropriate high or low alarm point.

RM03A - Rx Bldg North
RM03B - Rx Bldg South
RM03C - Tip Room

Removal of the malfunction will restore the selected area radiation monitor to normal operation.

REFERENCES:

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Failure to Auto Scram

NUMBER: RP 1

TYPE: Discrete

CAUSE: Short Circuit on All RPS Logic Channels (From 5A-F..A-D Fuse to 5A K13A-H Relays)

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will prevent the rods from inserting on receipt of an auto-scram signal. Indications will verify receipt of scram signal with appropriate annunciators. However, scram discharge volume level will not increase, rods will not move, and the full core display will indicate rods remaining at their position. Neutron flux will not decrease. Plant parameters will respond dynamically to the plant conditions that initiated the scram signal.

Manual scram and/or mode switch to shutdown capability is not affected by this malfunction. The reactor can be scrammed and the plant response will be normal.

Removal of the malfunction will restore the automatic scram functions.

REFERENCES: GEK 32430
CWD B-191301, sh. 803-828

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RPS MG (A,B) Failure

NUMBER: RP02

TYPE: Generic (A,B)

CAUSE: MG Set Motor Supply Breaker Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected RPS MG set will coastdown and power to the affected RPS bus will be lost, resulting in a half scram and half isolation (full isolation on PCIS group 3). The reactor building HVAC system will trip, standby gas treatment will start. Appropriate trip channels will actuate as will the corresponding logic channels in the power range neutron monitoring system, the off-gas, steam line radiation, and Rx building exhaust and refuel floor rad monitors. Appropriate annunciators will actuate for the selected RPS MG set power lost. All individual scram signal annunciators will alarm.

If a trip already exists in the alternate RPS channels, the reactor will scram. If both RPS MG sets failure malfunctions are active at the same time, the reactor will scram and a full isolation of PCIS groups 1-5 will occur. The affected RPS MG set bus may be connected to its alternate power supply.

If this alternate power supply is selected, power is restored and trips/annunciators may be reset.

Removal of the malfunction will allow the selected RPS MG set to be restarted via IDA-RPR01(2) to re-energize its bus, as long as the alternate power supply is not selected. Upon re-energization, all associated trips may be reset and annunciators cleared to restore affected logic channels to normal.

REFERENCES: OP 3134
GEK 32430
CWD B-191301, sh. 800-801

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Spurious Group I Isolation

NUMBER: RP03

TYPE: Discrete

CAUSE: Spurious Relay Failure, Such That Group I Isolation is Activated

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause a spurious Group I Isolation with no indication of cause. All MSIVs, main steam drain isolation valves, and recirculation loop sample line isolation valves will close. The reactor will scram due to MSIV position if the Reactor mode switch is in the run position. Safety relief valves will open to control reactor pressure. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore spurious relay failure and allow the isolation to be reset.

REFERENCES: CWD B-191301, sh. 1100 - 1103
GEK 32435

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Spurious Group II Isolation

NUMBER: RP04

TYPE: Discrete

CAUSE: Spurious Relay Failure Such That Group II Isolation is Activated

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause a spurious Group II isolation with no indication of cause. The following system isolation valves will close if open:

RHR discharge to radwaste (RHR-57,66)
Drywell floor drain (LRW-82,83)
Drywell equipment drain (LRW-94,95)
RHR Sample Valves FCV-161 + 160

All associated indication and annunciation will reflect the activation of the Group II isolation.

Removal of the malfunction will restore the spurious relay failure and allow the isolation to be reset.

REFERENCES: CWD B-191301, Sh. 1100-1103
GEK 32435

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Spurious Group III Isolation

NUMBER: RP05

TYPE: Disc:ete

CAUSE: Spurious Relay Failure Such That Group III Isolation is Activated

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause a spurious Group III isolation with no indication of cause. The following system isolation valves will close if open:

Drywell/torus air purge inlet (16-19-8,9)
Drywell purge and vent outlet (16-19-7A)
Drywell purge and vent outlet bypass (16-19-6A)
Drywell and torus exhaust to PTF-5 (16-19-7)
Drywell one-inch vents to SBGT VG-9A,VG-22A
Torus purge supply (16-19-10)
Torus purge and vent outlet (16-19-7B)
Torus purge and vent outlet bypass (16-19-6B)
Exhaust to SBGT system (16-19-6)
N2 purge supply valve (16-19-23)
Containment purge makeup (16-20-20,22A,22B)
Containment air compressor suction (CA-38A,38B)

Containment air sampling system (CAM-76A,B and VG-23,26)
Containment air dilution compressor discharge To containment (NG-11A,11B,12A,12B,13A,13B)
SBGT Starts
Reactor building HVAC (9,10,11,12) Isolate

All associated indications and annunciators will reflect the activation of the Group III isolation.

Removal of the malfunction will restore the spurious relay failure and allow the isolation to be reset.

REFERENCES: CWD B-191301, Sh. 1100-1103
GEK-32435

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Spurious Group IV Isolation

NUMBER: RP06

TYPE: Discrete

CAUSE: Spurious Relay Failure, Such That Group IV Isolation is Activated

PLANT CONDITIONS:

Reactor Shutdown/Cooldown and in Shutdown Cooling

EFFECTS: This malfunction will cause a spurious Group IV isolation with no indication of cause. The following system isolation valves will close if open:

RHR shutdown cooling supply (RHR-17,18)
RHR reactor head cooling (RHR-32,33)

All associated indications and annunciations will reflect the activation of the Group IV isolation. RHR pumps running in the shutdown cooling mode will trip upon loss of suction flow path.

Removal of malfunction will restore the spurious relay failure and allow the isolation to be reset.

REFERENCES: CWD B-191301, Sh. 1100-1103
GEK-32435

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: PCIS GROUP II (A,B) Isolation Failure

NUMBER: RP07

TYPE: Generic (A,B)

CAUSE: Isolation Relay(s) Stuck
16A-K17, 16A-K18 Relays

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected PCIS relay will fail to operate as required by the corresponding trip logic. The resulting relay failure will cause those valves associated with the group II PCIS I or PCIS II system valves from automatically isolating if open. The affected valves will operate using each associated control switch. No annunciators will be affected by this failure

If a trip already exists, once the malfunction is removed the associated system will activate causing an isolation of the affected valves.

REFERENCES: OP 2115
CWD B-191301, sh. 1112-1116

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: PCIS GROUP III (A,B) Isolation Failure

NUMBER: RP08

TYPE: Generic (A,B)

CAUSE: Isolation Relay(s) Stuck
16A-K23, 16A-K24 Relays

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected PCIS relay will fail to operate as required by the corresponding trip logic. The resulting relay failure will cause those valves associated with the group III PCIS I or PCIS II system valves from automatically isolating if open. The affected valves will operate using each associated control switch. No annunciators will be affected by this failure

If a trip already exists, once the malfunction is removed the associated system will activate causing an isolation of the affected valves.

REFERENCES: OP 2115
CWD B-191301, sh. 1112-1116

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: PCIS GROUP V (A,B) Isolation Failure

NUMBER: RP09

TYPE: Generic (A,B)

CAUSE: Isolation Relay(s) Stuck
16A-K26, 16A-K27 Relays

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: The selected PCIS relay will fail to operate as required by the corresponding trip logic. The resulting relay failure will cause those valves associated with the group V PCIS I or PCIS II system valves from automatically isolating if open. The affected valves will operate using associated control switches. No annunciators will be affected by this failure

If a trip already exists, once the malfunction is removed the associated system will activate causing an isolation of the affected valves.

REFERENCES: OP 2115
CWD B-191301, sh. 912

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Loop (A,B) Rupture

NUMBER: RR01

TYPE: Generic, Variable (100% Severity = Complete Break At Normal 100% Design Pressure, 28 inch Pipe)

CAUSE: Pipe Failure

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in a rupture of the selected recirc loop (A,B) due to pipe failure at the pump suction line. A loss of mass inventory from the reactor vessel to the primary containment will occur dependent on severity selected. The reactor will scram due to either reactor low level or high drywell pressure with a subsequent isolation signal. Flow in the affected loop will go to zero immediately, total drive flow will be a function of pump coast down characteristics of the remaining recirculation pump. As level decreases, the jet pump suctions will uncover resulting in core flow dropping to zero. The response of the feedwater system will be of little affect.

Appropriate ECCS equipment will start and resupply coolant to the reactor vessel. Vessel reflood will be a function of emergency core cooling systems operating. Drywell temperature and pressure increase will be a function of the energy addition.

Removal of the malfunction requires reinitialization of the Simulator.

REFERENCES: FSAR, Section 14.6.3

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Temperature Equalizing Column Failure (A,B)

NUMBER: RR02

TYPE: Generic (A,B)

CAUSE: Steam Impinging on Reactor Level Fixed Reference Leg Located Upstream of Drywell Penetration X-280 (X-290D)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected reactor temperature equaling column to fail due to steam impinging on the associated reactor water level fixed reference leg located upstream of drywell penetration X-280 (X-290D). The net effect of increased temperature will be reflected on all of the following instrumentation erroneously indicating actual level:

A) LI-72A (PNL 9-5)
B) LI-278 (PNL-9-5)
C) LT-72A,LT72C
D) LT-72B,LT-72D
E) LT-57A,LT-57B
F) LT-58A,LT-58B,LT-70,LT-6-52A,(B)
G) LI-57A,LI-57B (PNL 9-5)
LI-86 on panel 9-4 will not be affected.
H) LI-91A,LI-91B (PNL 9-3)

Applicable alarms and trip setpoints associated with the systems initiated by the reactor protection or safeguard system level switches will respond as required. Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191267, 191167

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Jet Pump Failure

NUMBER: RR03

TYPE: Generic, A(JP1+2) through J(JP19+20)

CAUSE: Riser Break

PLANT CONDITIONS:

Reactor Startup or Power Operating Condition

EFFECTS: This malfunction will result in selected recirc jet pump [A(JP1+2) through J(JP19+20)] failure due to inlet riser break internal to the RPV just below the top allowing the entire rams head assembly to become unattached. Recirculation flow will discharge directly into the downcomer region. The flow distribution of the affected loop will be changed causing reduced flow to all other affected loop jet pumps. (Example - JP1 and JP2) will drop to zero and increase as reverse flow occurs from the lower plenum up into the downcomer. The reduced core flow will result in a noticeable decrease in reactor power.

Removal of malfunction will require reinitialization of the Simulator.

REFERENCES: P+ID G-191267
GEK 9609 Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Erratic Recirc Jet Pump Flow

NUMBER: RR04

TYPE: Generic A(JP1+2) through J(JP19+20)

CAUSE: Loose Ramshead

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in selected erratic recirc jet pump [A(JP1+2) through J(JP19+20)] flow due to loose ramshead with subsequent change in orientation. The misdirected flow causes random oscillation about the normal value for selected jet pump flow. No significant effects will be evident in overall loop flow or core flow.

Removal of the malfunction will require simulator reinitialization.

REFERENCES: P+ID G-191267
GEK 9609, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Drive Motor Breaker (A,B) Trip

NUMBER: RR05

TYPE: Generic (A,B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected recirc motor-generator set drive motor breaker (A,B) to trip due to an overload trip. Drive motor current will spike and then drop to zero and the associated motor generator supply breaker and field breaker will trip. The "DRIVE MOTOR (A,B) TRIP", "GEN A(B) Lockout" annunciator will actuate. The associated reactor recirculation pump will coast down with subsequent decrease in loop flow. The reduction in core flow will cause an abrupt decrease in core power to approximately 60%. Flow through the affected loop jet pumps will coast down to zero and then increase slightly as reverse flow occurs.

Removal of the malfunction will return the selected drive motor breaker to normal. Use of IDA to rest lockouts must be used to reset the lockout relays.

REFERENCES: GEK 9609

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Pump Motor-Generator Field Breaker Trip

NUMBER: RR06

TYPE: Generic (A,B)

CAUSE: Overload Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected Rx recirc motor-generator A,(B) field breaker to trip due to overload trip. The net effects will be the same as malfunction RR05 (Drive motor breaker trip) except the affected field breaker trips simultaneously as the drive motor trips. This produces a faster coastdown time due to the loss of inertia with drive motor.

Removal of malfunction will restore selected Rx recirc M-G set field breaker to normal. Use of IDA to reset lockouts must be used to reset lockout relays.

REFERENCES: GEK 9609 Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Pump (A,B) Lower Seal Failure

NUMBER: RR07

TYPE: Generic, Variable (A,B)

CAUSE: Worn Seal (100% Severity = 1.1GMP)
NOTE: For Failure of Both Upper and Lower Seal, 100% = 20GPM

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the selected recirc pump A(B) lower seal failure due to worn seals. (100% severity = 1.1 GPM) Failure of No. 1 seal (lower seal) is characterized by No. 2 seal (upper seal) pressure approaching No. 1 seal pressure with the seal leakage through No. 2 orifice increasing to about 1.1 GPM at 100% severity giving a high controlled leakage alarm on CRP 9-4. Seal temperatures will decrease with the increase in leakage. Drywell equipment drain sump pumping rate will increase as well as drywell air temperature and pressure.

If malfunction RR08A(B) is active, leakage will be limited to 20 GPM at 100% severity.

Removal of malfunction will require reinitialization of the Simulator.

REFERENCES: GEK 9609, Reactor Recirculation
OP 2110
P+ID G-191159

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Pump (A,B) Upper Seal Failure

NUMBER: RR08

TYPE: Generic, Variable (A,B)

CAUSE: Worn Seal (100% Severity = 1.1 GPM)
NOTE: For Failure of Both Upper and Lower Seal, 100% = 20 GPM

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the selected recirc pump A(B) upper seal failure due to worn seals. (100% severity = 1.1 GPM) failure of No. 2 seal (upper seal) is characterized by No. 2 seal pressure dropping with concurrent high seal leakage alarm. No. 1 seal (lower seal) will not be affected. Seal temperatures will decrease with the increase in leakage. Drywell air temperatures and pressure will increase accordingly.

If malfunction RR07A(B) is active concurrently leakage will be limited to 20 GPM at 100% severity.

Removal of malfunction will require reinitialization of the simulator.

REFERENCES: GEK 9609, Reactor Recirculation
OP 2110

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirculation Pump (A,B) Locked Rotor

NUMBER: RR09

TYPE: Generic (A,B)

CAUSE: Mechanical Seizure of Pump Shaft

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected Rx recirc pump rotor, (A,B) to lock up due to mechanical seizure of pump shaft. The affected recirc pump will stop suddenly. Pump motor current and power will increase to starting values. The affected Rx recirc pump `PP MTR A(B) LOCKED ROTOR TRIP' will cause actuation of the generator lockout relay which in turn trips the associated MG drive motor breaker and field breaker. Pump motor temperature will increase for several minutes following the pump trip then begin to slowly decrease to normal over 15 minutes. Loop drive flow will drop to zero in less than 1.0 second. Drive motor amp increase will not be observable due to the speed of the transient.

Total core flow will decrease proportionally resulting in increased voiding and a rapid power reduction. Vessel level will swell and will reach the high level trip conditions with subsequent closure of the main turbine stop valves, tripping of RCIC and HPCI. Rx will scram from TSV closure.

Removal of the malfunction will restore the affected recirc pump rotor to normal. Use of IDA to reset lockout relay will permit pump restart.

REFERENCES: GEK 9609 Reactor Recirculation
OP 2110 Reactor Recirculation
P+ID G-191159

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Master Flow Controller Failure

NUMBER: RR10

TYPE: Discrete, Variable (0-100% Controller Output)

CAUSE: Electronic Failure of Master Controller Output

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: This malfunction will result in the failure of the Rx recirc master flow controller due to electronic failure of controller output. The output of the master flow controller will go to the instructor specified value. The individual loop speed controllers will respond to the changing input signal and adjust the associated Rx recirc pump MG speed accordingly. The input to the loop side speed controllers will be limited to the range of approximately 50-102.5%, regardless of master controller output.

The changing recirculation pump speed will cause reactor power to change. However, a high power scram is not expected to occur. If the loop speed controllers are placed in manual, the individual recirc pump speeds can be returned to normal.

Removal of the malfunction will allow the master flow controller output to return to its normal value.

REFERENCES: GEK 9609, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMUL‹ .OR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Individual Loop (A,B) Flow Controller Failure

NUMBER: RR11

TYPE: Generic, Variable (A,B) (0-100% Controller Output)

CAUSE: Electronic Failure of Individual Controller Output

PLANT CONDITIONS:

Reactor Startup or Operating at Power

EFFECTS: This malfunction will result in the failure of the selected (A,B) Rx recirc. Individual loop flow controller due to electronic failure of the affected controller output. The output of the selected individual loop speed controller will go to the instructor specified value. If the loop speed controller is in the `Balance' mode the deviation meter will move in the opposite direction. The recirc pump speed will respond accordingly. If feedwater flow i- less than 20%, the output of the loop flow controller will not be limited. but the input to the basic speed controller will be limited to 20%.

This malfunction will affect both auto and manual modes of operation of individual loop flow controller.

Removal of the malfunction will allow the output of the selected loop flow controller to return to normal.

REFERENCES: GEK 96C9, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc MG Set (A,B) Incomplete Sequence

NUMBER: RR12

TYPE: Generic (A,B)

CAUSE: Relay Failure Prevents Field Breaker Closure

PLANT CONDITIONS:

Recirc Pump Start

EFFECTS: This malfunction will allow the selected Rx recirc loop (A,B) pump to be started, however the pump start sequence timer will initiate an incomplete startup sequence due to relay failure with subsequent field breaker not closing within approximately 16 seconds.

The following sequence of events will occur:

A. Alarm 'MG Set A(B) Seq Incom' actuates
B. Alarm 'GEN A(B) LKOT' actuates
C. Alarm 'MG Set A(B) SCP Tube LOK' actuates
D. Alarm 'DRV MTR 2 Trip' actuates
E. Drive Motor Trips
F. Generator trips and is locked out field breaker does not close)
G. Rx recirc pump parameters return to initial prestart values

The selected Rx recirc loop may be started again after all associated interlocks have been reset, however, the above events will be repeated. If the Rx recirc loop is in operation upon malfunction activation, the pump will not trip.

Removal of the malfunction will restore failed relay to normal, allowing normal start sequence completion. Use of IDA for resetting GEN lockout devices will be required.

REFERENCES: GEK - Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc Pump Scoop Tube (A,B) Lockup

NUMBER: RR13

TYPE: Generic, A,B

CAUSE: Loss of Control Power

PLANT CONDITIONS:

Recirc Pump Operating

EFFECTS: This malfunction will cause the selected Rx recirc pump (A,B) scoop tube to lock up due to loss of control power. The following events occur as a result of the failure.

A. The scoop tube on the affected recirc pump MG set will lock at its present position.

B. Alarm 'MG set A(B) SCP tube LOK' actuates

C. The selected recirc pump will not respond to any changes in the speed control signal, manually or automatically.

Removal of the malfunction will restore power to the scoop tube and allow the scoop tube lockup reset pushbutton to reset the scoop tube with subsequent response from the speed control as required.

REFERENCES: GEK - 9609, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc Pump (A,B) Discharge Valve Gate Separation

NUMBER: RR14

TYPE: Generic (A,B)

CAUSE: Valve Failure (Malfunction Will Only Actuate When Valve is Fully Closed)

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will result in the selected Rx recirc pump discharge valve V-2-53A,(V-2-53B) gate to separate due to mechanical failure of the valve. This malfunction will only be actuated when the selected valve is in the fully closed position. Selected pump discharge valve position indication will not be affected. Operation of the control room panel (9-4) valve control switch will not be affected. However, the affected loop flow will not increase as expected from normal valve opening. Recirc loop flow will be limited to capacity of pump discharge bypass valve V-2-54A(B).

Removal of the malfunction will require reinitialization of the Simulator.

REFERENCES: GEK 9609, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Yarway Variable Leg (A,B) Draining (Instrument Line Failure Outside of Drywell)

NUMBER: RR15

TYPE: Generic (A,B)

CAUSE: Piping Failure of Nuclear Boiler Vessel Instrumentation Outside of Drywell Located Between Root Valve 14A(B) and SL-15A(B)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the failure of the selected yarway variable leg (A,B) for the nuclear boiler vessel instrumentation outside of the primary containment located between root valve 14A(B) and SL-15A(B), due to piping failure. This event results in a 100 GPM leak from the reactor pressure vessel to the secondary containment. All affected level instrumentation will indicate fully downscale. Water level logic A(B) will indicate the tripped condition. Area radiation monitoring will indicate an increase in radiation level of the area. The piping failure may be isolated by use of IDA for instrument root valve. Removal of the malfunction will return the yarway variable leg and vessel instrumentation to normal.

REFERENCES: P+ID G-191267

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc Pump (A,B) Discharge Valve Fails to Open

NUMBER: RR16

TYPE: Generic, (A,B)

CAUSE: Mechanical Binding

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will result in the selected Rx recirc pump discharge valve V-2-5A(B) failing to open due to mechanical binding. Selected pump discharge valve position indication will reflect this event. The affected Rx recirc loop flow cannot be changed except from operation of the bypass valve.

Removal of the malfunction will return the selected Rx recirc pump discharge valve to normal allowing operation of the valve.

REFERENCES: GEK 9609 - Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Recirc MG AC Lube Oil Pump Trip

NUMBER: RR17

TYPE: Generic, (A,B,C,D,E,F)

CAUSE: Overload Trip

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected Rx recirc MG set AC lube oil pump (A-F) to trip if running. The standby AC lube oil pump will auto start on interlock. Loss of all AC lube oil pumps will result in the auto start of the DC lube oil pump at 10 PSIG, the associated Rx recirc MG set drive motor breaker will trip due to "MG set A(B) L.O.PRLO, a scoop tube lock will occur.

Removal of the malfunction will allow return selected AC lube oil pump to normal.

REFERENCES: GEK 9609, Reactor Recirculation

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Rod Worch Minimizer Failure

NUMBER: RW01

TYPE: Discrete

CAUSE: Failure of the RWM Automatic Bypass Circuit

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the RWM to become bypassed regardless of power level. The auto bypass status light will actuate. Any withdraw or insert errors will not produce a rod block.

Removal of the malfunction will allow the RWM to operate as required.

REFERENCES: GEK 34687

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Rod Worth Minimizer Fails to Initialize

NUMBER: RW02

TYPE: Discrete

CAUSE: RWM Computer Failure

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will cause the Rod Worth Minimizer to fail to initialize due to RWM computer failure. The RWM will not allow latching the preprogrammed control rod withdrawal/insertion sequence A(B). A RWM rod block will exist prohibiting control rod withdrawal movement.

Removal of the malfunction will allow the Rod Worth Minimizer to initialize as a required and subsequently latching the selected rod sequence.

REFERENCES: GEK 34687, Rod Worth Minimizer

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Fuel Clad Failure

NUMBER: RX01

TYPE: Discrete, Variable (100% = 5.5×10^4 R/hr)

CAUSE: Fuel Clad Degradation

PLANT CONDITIONS:

Reactor Operating at Power, 1 - 100%

EFFECTS: This malfunction will cause an increase in coolant activity at the selected severity in addition to the normal full power background level. The increased activity will be evidenced on the main steam line radiation monitors. Fission product activity released by the fuel cladding failure will be propagated from the reactor throughout the plant by main steam and/or other systems receiving water or steam from this source. Radiation alarms and indications will respond appropriately. As severity increases to MSIV closure trip setpoint, a reactor scram occurs as well as a Group 1 isolation. Reactor pressure will increase as a function of initial power level and decay heat generation. Relief valves will operate as necessary.

Removal of the malfunction will require re-initialization of the Simulator.

REFERENCES: FSAR, Section 14.6.3

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Increased Rod (XX-YY) Worth

NUMBER: RX02

TYPE: Generic

CAUSE: Selected Control Rod (XX-YY) Worth Increases 200%

PLANT CONDITIONS:

Plant Startup or Power Operation

EFFECTS: This malfunction will cause the selected control rod(s) (XX-YY) worth to increase 200%, when it is withdrawn to the new position. The associated reactivity change to the core will be reflected on all neutron monitoring instrumentation. Reinsertion of the affected control rod(s) will return reactivity to its original value, if the upscale trip setpoint did not cause a reactor scram, all associated alarms will actuate as the event occurs.

Removal of the malfunction for the selected control rod will return that rod's worth to normal expected values.

REFERENCES: FSAR, Section 14

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SLC Pump (A,B) Trip

NUMBER: SL01

TYPE: Generic (A,B)

CAUSE: Relay (42) Failure

PLANT CONDITIONS:

SLC System Operating

EFFECTS: This malfunction will cause the selected SLC pump(s) to trip due to a failure of relay (42) device, for the affected motor. The SLC pump discharge header pressure will reflect actual reactor pressure. The affected SLC pump red (running) indicating lamp will extinguish and green (stop) lamp will illuminate on CRP 9-5. If SLC is being injected to the RPV at the time of trip, neutron activity response will depend on core reactivity. SLC tank level will stop decreasing. The red flow indicating lamp will extinguish. The operator may use the unaffected SLC pump to continue the boron solution injection. All applicable annunciators will actuate when appropriate.

Removal of the malfunction will restore SLC pump(s) to normal operation.

REFERENCES: CWD B-191301, sh. 1200,1203
P+ID G-191171
OP 2124

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SLC Squib Valve (A,B) Fails to Fire

NUMBER: SL02

TYPE: Generic (A,B)

CAUSE: Switch 11A-S1 Contacts 5-5C and/or 6-6C Fail Open

PLANT CONDITIONS:

SLC System Operation Required

EFFECTS: This malfunction will cause the selected SLC squib valve(s) not to fire. The continuity indicating lamp(s) will remain illuminated.

If the standby liquid control switch is placed in system 1 position, the pump will start but the squib valve will not open. Pump discharge pressure will rise to about 1400 psig. The relief valve will open allowing the pump discharge to return to SLC tank. No transient will be observable on the tank level indicator. The RWCU system will isolate, but no boron solution will reach the reactor vessel. SLC (red) flow indicating lamp will remain extinguished.

Removal of the malfunction will restore the failed contact(s) to normal. If the SLC switch is still in the system 1 position, the squib valve will fire and boron solution will begin to be pumped into the reactor.

REFERENCES: CWD B-191301, sh. 1200-1203
P+ID G-191171
OP 2124

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: SLC Storage Tank Leak

NUMBER: SL03

TYPE: Discrete, Variable (100% = 50% Pump Suction Pipe Diameter, 2 1/2 inch)

CAUSE: SLC Pump A Suction Flange Leak

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause SLC pump "A" suction flange to leak at severity selected. Boron solution will leak too the SLC compartment floor. SLC tank level will decrease, the rate being dependent upon severity selected.

If SLC flange is allowed to continue to leak, tank level will decrease to zero. SLC low level annunciator will actuate and tank temperature will decrease to ambient more rapidly since the detector is uncovered. The low temperature annunciator will actuate if temperature decreases below 75°F.

If the pumps are running with no solution in the tank, or are running when level reaches zero, discharge pressure will decrease as if the pump had been stopped, dropping quickly to reactor pressure and then decaying to atmospheric pressure.

Removal of the malfunction will stop the leakage. If desired, the tank may be refilled using an IDA.

REFERENCES: CWD B-191301, sh. 1200-1203
P+ID G-191171
OP 2124

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW Pump (A,B) Trip

NUMBER: SW01

TYPE: Generic (A,B)

CAUSE: Activation of Magnetic Overload Device

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected RBCCW pump to trip. RBCCW system flow and pressure will decrease. As system pressure decreases to approximately 74 psig the standby pump will auto start, providing its control switch is not in the pull-to-lock position. System flow and pressure will return to normal.

If both RBCCW pumps are tripped the temperatures of equipment cooled by RBCCW will increase. These components include: recirc pump coolers, RRUs, drywell equipment drain sump cooler, fuel pool heat exchangers, NRHX, RWCU pump coolers, RHR pump coolers, CRD pump coolers, reactor building drain sump coolers, containment air compressor and radwaste equipment. All associated annunciators will actuate when appropriate.

Removal of malfunction(s) will allow restart of the affected pump(s) and subsequent decay of high temperatures on associated equipment.

REFERENCES: CWD B-191301, sh. 441-442
OP 2112
P+ID G-191159, sh. 3

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of RBCCW Flow to Drywell Coolers (RRU-1,2,3,4)

NUMBER: SW02

TYPE: Generic (RRU-1,2,3,4)

CAUSE: Selected RBCCW Outlet Valve(s) RCW-71C, 72C, 73C, 74C Fails Closed

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause selected RRU RBCCW outlet valve(s) to close, resulting in a loss of flow to the drywell cooler(s). The loss of flow through each cooler will be indicated by an increase in air outlet temperature on recorder TRI-149-1 CRP 9-25.

The loss of RBCCW flow to the drywell coolers will result in an increase in drywell temperature and pressure. The magnitude of the temperature and pressure increase will depend upon the number of coolers affected. High drywell pressure will result in a reactor scram, isolation signal and ECCS actuation. Associated annunciators will actuate when appropriate.

Removal of the malfunction will restore RRU RBCCW outlet valve(s) to normal.

REFERENCES: P+ID G-191159, sh. 3
CWD B-191301, sh. 1226, 1415-1418
OP 2182

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RWCU NRHX Temperature Control Valve (TCV-5A) Fails Closed

NUMBER: SW03

TYPE: Discrete

CAUSE: Device TC-104-5 Fails, Such That (TCV-5A) Closes

PLANT CONDITIONS:

RWCU System Operating

EFFECTS: This malfunction will cause TC-104-5 to fail, which closes TCV-5A. RBCCW flow to RWCU non-regenerative heat exchangers will stop. Clean up filter inlet temperature high annunciator will actuate at 130°F. RWCU system will isolate at 140°F., the system supply isolation valves MOV15 and 18 and system return valve MOV68 auto close simultaneously. The RWCU pumps trip due to isolation valves not full open. Clean up heat exchanger temperature high annunciator will actuate at 140°F.

Removal of the malfunction will return TC-104-5 to normal operation.

REFERENCES: P+ID G-191159, sh. 3
G-191178, sh. 1
CWD B-191301, sh. 903-904, 909-912, 923, 1121
OP 2112

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW Surge Tank Makeup Valve (LCV-1) Failure

NUMBER: SW04

TYPE: Discrete, Variable (0 - 100% of Valve Position)

CAUSE: Solenoid Valve Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the RBCCW surge tank makeup valve to fail at the severity selected from 0 to 100% of valve position.

At 100% severity the makeup valve will fail to the full open position causing the surge tank to fill up and overflow to the reactor building floor drain system sump. Makeup water to the RBCCW surge tank can be isolated by closing RCW-142A via IDA.

At 0% severity the makeup valve will fail to the full closed position preventing any makeup to the surge tank to make up for the water lost due to leakage. Depending on the size of the leakage, the surge tank and surge line will eventually empty which will result in RBCCW pump cavitation. Pump cavitation will result in a loss of system flow and eventual pump trip on overcurrent due to binding of the pump wearing rings.

Removal of the malfunction will restore the solenoid valve to normal.

REFERENCES: P+ID G-191159, sh. 3
OP 2182
CWD B-191301, sh. 463

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW Heat Exchanger Tube Leak

NUMBER: SW05

TYPE: Generic (A,B) Variable (100% = 50 GPM at Normal Differential Pressure)

CAUSE: RBCCW Heat Exchanger Tube Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected RBCCW HX(s) tubes to leak service water into the RBCCW System. RBCCW surge tank level will increase. Dependent upon severity of leak selected will determine the time the high level annunciator in PNL 9-6 will actuate. If no action is taken to isolate the leaking heat exchanger, the surge tank will overflow to the reactor building floor and be collected by the reactor building floor drain system sump. Appropriate annunciators will actuate.

The affected heat exchanger may be isolated using the IDA function provided. The malfunction can be removed restoring the RBCCW heat exchanger to normal.

REFERENCES: OP 2182
P+ID G-191159, sh. 3
CWD B-191301, sh. 54

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RBCCW System Header Leak

NUMBER: SW06

TYPE: Discrete, Variable (100% = 50% Pipe Diameter 8 inch Pipe)

CAUSE: Pipe Failure at Discharge of the RBCCW Heat Exchanger Downstream of RCW-118.

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in leakage of RBCCW system water into the reactor building. The water will collect in the reactor building floor drain sump. If level reaches the high level setpoint, a sump high level alarm will actuate.

If leakage out of the RBCCW system is equal to or less than the surge tank makeup flow rate, the RBCCW system inventory will be maintained. If the RBCCW system inventory cannot be maintained, the surge tank level will decrease. The rate of decrease will be dependent upon the malfunction severity.

When the pump suction pressure decreases below the required NPSH, the RBCCW pumps will cavitate. Additional loss of inventory will result in a loss of system flow and eventual pump trip.

Removal of the malfunction will stop the leak.

REFERENCES: P+ID G-191159, sh. 3
OP 2182

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Water Pump (A,B,C,D) Trip

NUMBER: SW07

TYPE: Generic (A,B,C,D)

CAUSE: Activation of Instantaneous Overcurrent Device (50)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in the selected service water pump to trip, if running. The associated pump trip annunciator will actuate. System flow and pressure will begin to decrease. If pressure approaches approximately 90 psig, the standby pump will auto start as required by the control switch. Heat load temperatures will be dependent upon the flow through the system.

Removal of the malfunction will restore the affected service water pump to normal operation.

REFERENCES: P+ID G-191159
OP 2181
CWD B-191301, sh. 424-427

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: RHR Service Water Pump (A,B,C,D) Trip

NUMBER: SW08

TYPE: Generic (A,B,C,D)

CAUSE: Activation of Instantaneous Overcurrent Device (50)

PLANT CONDITIONS:

RHR Service Water System Operating

EFFECTS: This malfunction will cause the selected RHR Service Water pump, if running, to trip. The associated pump trip annunciator will actuate. The affected loop flow will decrease to zero and pressure will go to static pressure. If the alternate pump is operating, flow and pressure will drop to the capacity of the remaining pump.

During shutdown cooling operations, the reduction in heat removal will reflect back to the RHR system and reactor coolant system. If the heat exchanger is being utilized for suppression pool cooling, the loss of heat removal will be seen in suppression pool temperature.

Removal of the malfunction will allow the affected pump to be restarted.

REFERENCES: F+ID G-191159, 191172
CWD B-191301, sh. 1304-1307

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Water Crossconnect Valve (SW-19A/B) Fails Closed

NUMBER: SW09

TYPE: Generic (A,B)

CAUSE: Switch SW-90A Closing Contacts (3-3T) Weld Closed

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected service water crossconnect valve SW-19A and/or 19B, if open, to fail to the closed position. If valve is already closed, it will not open. Valve position indicating light on CRP 9-6 will reflect the closure of the selected valve. No other effects will be seen.

Removal of the malfunction will restore service water crossconnect valve(s) to normal operation.

REFERENCES: P+ID G-191159
CWD B-191301, sh. 458A,B

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Water Strainer (A,B) Plugged

NUMBER: SW10

TYPE: Generic (A,B), Variable (100% = Complete Blockage)

CAUSE: Marine Growth

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in blockage of the selected service water strainer. The blockage rate will be limited to 10% per minute, independent of the malfunction severity ramp time. The blockage will result in reduced flow through the strainer and an increase in strainer differential pressure. As strainer differential pressure increases, a high D/P alarm will actuate at 4 psid.

If only one service water header is in operation, service water header flow and pressure will decrease. Heat load temperatures of the systems supplied by service water will be dependent on the flow through the system.

The strainer bypass valve may be operated at any time using an IDA to restore system flow.

Removal of the malfunction will return the strainer to normal operation.

REFERENCES: P+ID G-191159
OP 2191

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Water Traveling Screen (A,B) Fouling

NUMBER: SW11

TYPE: Generic (A,B), Variable (100% = Service Water Pump Trip)

CAUSE: Debris in Service Water Intake

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected service water traveling screen to foul to the desired severity. Screen wash will have no effect on the rate at which the screen diff level increases. Traveling screen diff high annunciator will actuate at 10 inches water. At 100% severity, service water system pressure and flow will decrease and in approximately 10 mins. The affected service water pump(s) will cavitate and trip on overcurrent. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore the affected service water traveling screen to normal operation, however, if the overcurrent trip occurs due to cavitation, removal of malfunction will require Simulator reinitialization.

REFERENCES: P+ID G-191159

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Service Water Pipe Failure in Turbine Building

NUMBER: SW12

TYPE: Discrete, Variable (100% = Complete Pipe Break, 16 inch pipe)

CAUSE: Flange Break at Spool Piece "B" Off Nonessential Service Water Header

PLANT CONDITIONS:

Service Water System Operating

EFFECTS: This malfunction will cause service water to leak, at the severity selected, from spool piece "B" flange to the turbine building floor sump system. Service water system pressure decreases. Service water header pressure low annunciator on CRP 9-6 actuates. The standby service water pump(s), if in automatic, will start. Heat load temperatures of the systems serviced by service water will be dependent on the flow through each individual component. Changes in system flow will be indicated on the pump current meters. All associated annunciators will actuate when appropriate.

Closing turbine building isolation valve SW-20 will stop the leak and isolate the nonessential loads. Immediate shutdown of the plant would be required.

Removal of the malfunction will return the service water system to normal operation.

REFERENCES: P+ID G-191159, sh. 2

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Loss of Service Water to the Emergency Diesel Generator

NUMBER: SW13

TYPE: Generic (A,B)

CAUSE: Diesel Generator Service Water FCV28A/B Solenoid Fails to the Closed Position

PLANT CONDITIONS:

Emergency Diesel Generator Running

EFFECTS: This malfunction will cause the selected diesel generator service water flow control valve to close or if already closed it will not open. This will result in a loss of cooling water to the diesel generator, which will cause diesel generator temperatures to increase. The diesel will eventually trip due to high temperature. All associated annunciators will actuate when appropriate.

Removal of the malfunction will restore the diesel generator service water FCV28A and/or 28B to normal.

REFERENCES: CWD B-191301, sh. 610-611
P+ID G-191159, 191270
Fairbanks Morse Diesel Generator Technical Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: TBCCW Pump (A,B) Trip

NUMBER: SW14

TYPE: Generic (A,B)

CAUSE: Activation of Thermal Relay Device (49)

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected TBCCW pump to trip. If TBCCW system flow and pressure decreases to 65 psig the standby pump will auto start, providing its control switch is not in the pull-to-lock position. System pressure and flow will return to their original value.

If both TBCCW pumps are tripped, the associated temperatures of equipment cooled by TBCCW will begin to increase.

Removal of the malfunction will allow restart of the affected pump(s) and subsequent decay of high temperatures of associated equipment.

REFERENCES: CWD B-191301, sh. 456-457
OP 2183
P+ID G-191159, sh.4

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: TBCCW Heat Exchanger (A,B) Tube Leak

NUMBER: SW15

TYPE: Generic (A,B), Variable (100% = 50 GPM at Normal Differential Pressure)

CAUSE: Selected A,B TBCCW Heat Exchanger Tube Failure

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will cause the selected TBCCW HX(s) tubes to leak service water into the TBCCW system. TBCCW surge tank level will increase. Dependent upon severity of leak selected will determine the time the high level annunciator on CRP 9-6 will actuate. If no action is taken to isolate the leaking heat exchanger the surge tank will overflow to the floor and be collected by the turbine building floor drain system. Appropriate annunciators will actuate. The heat exchanger may be isolated via IDA.

Removal of the malfunction will restore the TBCCW heat exchanger to normal.

REFERENCES: P+ID G-191159, sh. 4
RP 2183
CWD B-191301, sh. 54

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: TBCCW Heat Exchanger (A,B) Relief Valve Lift

NUMBER: SW16

TYPE: Generic (A,B)

CAUSE: Faulty TBCCW Relief Valve

PLANT CONDITIONS:

Any Plant Condition

EFFECTS: This malfunction will result in leakage of TBCCW system water into the turbine building. The water will collect in the turbine building floor drain system sump. If the sump level reaches the high level setpoint, a sump high level alarm will actuate.

If the leakage out of the TBCCW system is equal to or less than the surge tank makeup flow rate, the TBCCW system inventory will be maintained. If the TBCCW system inventory cannot be maintained, the surge tank level will decrease. The rate of decrease will depend upon the malfunction severity.

When the pump suction pressure decreases below the required NPSH, the TBCCW pumps will start to cavitate. Additional loss of inventory will result in a loss of system flow and eventual pump trip on overcurrent due to binding of the pump wearing rings. The effects of loss of system flow are described in malfunction SW14 effects.

Removal of the malfunction will stop the leak.

REFERENCES: P+ID G-191159, sh. 4
RP 2183

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Trip

NUMBER: TC01

TYPE: Discrete

CAUSE: Emergency Governor Trip Valve Latch Failure

PLANT CONDITIONS:

Reactor Operating At Power, Turbine Operating

EFFECTS: The emergency trip valve will shift causing the turbine to trip. The overspeed trip indicating light and annunciator will actuate. The reactor will scram as the turbine stop valves close if turbine first stage press is $\geq$ 220 psig. The bypass valves will open to control pressure. Generator output will drop rapidly to zero and the generator will trip due to a Gen lockout caused by the turbine trip.

Attempting to reset the master trip will cause the trip indicating light to extinguish and the annunciator to clear, but the device will trip again when the switches returned.

Removal of this malfunction will restore normal operation of the emergency trip valve mechanism.

REFERENCES: GE Turbine Technical Manual
CWD B-191301, sh. 133

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Bypass Valve (1-10) Fails Open

NUMBER: TC02

TYPE: Generic (1,2,3,4,5,6,7,8,9,10)

CAUSE: Valve Servo Motor Failure

PLANT CONDITIONS:

Plant Startup/Shutdown or Power Operation

EFFECTS: This malfunction will cause the selected turbine bypass valve(s) to fail open. The affected bypass valve(s) will go to the full open position if closed or will remain open if required to close. If EPR/MPR in control the MHC system will respond to maintain reactor pressure constant. If the pressure control system cannot maintain pressure, the reactor will depressurize and if < 800 psig reactor pressure is reached with the mode switch in run a Group I isolation will occur. If a scram occurs and bypass valves are failed open, reactor cooldown may be excessive.

Removal of the malfunction(s) will return the turbine bypass valve(s) to normal operations.

REFERENCES: GEK 5585

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Bypass Valve (1-10) Fails Closed

NUMBER: TC03

TYPE: Generic (1,2,3,4,5,6,7,8,9,10)

CAUSE: Valve Servo Motor Failure

PLANT CONDITIONS:

Plant Startup/Shutdown or Power Operation

EFFECTS: This malfunction will result in the selected bypass valve(s) to go to the full closed position if open or will remain in the closed position if required to open. Reactor main steam pressure control system will respond as necessary to maintain reactor pressure constant. If the bypass valves are being used to maintain reactor pressure, pressure will begin to increase and the noneffected bypass valves will maintain pressure. If at higher power levels, safety/relief valves will relieve reactor pressure when the appropriate valve setpoints are reached. Torus temperature and levels will reflect this energy addition. All associated indicators, recorders and alarms will respond as appropriate.

Removal of the malfunction will return the turbine bypass valve(s) affected to normal operation.

REFERENCES: GEK 5585

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Pressure Regulator Oscillation (EPR,MPR)

NUMBER: TC04

TYPE: Discrete, Variable (100%=Plus and Minus 10 psig Swing) (A=EPR, B=MPR)

CAUSE: Control System Failure

PLANT CONDITIONS:

Reactor Operating at Power, 1-100%

EFFECTS: This malfunction will cause the selected pressure regulator (EPR,MPR) to oscillate if controlling reactor pressure at the time of the malfunction insertion. This will cause variations in turbine main control valves (or bypass valves) positions. As the control valves increase and decrease position, they will cause subsequent perturbations in reactor pressure and main steam flow/feedwater flow. The net effect will be reactor power oscillations, and possible reactor scram on high neutron flux also vessel level swing with the steam/feed flow changes. Indications of valve position, reactor vessel level, power, steam flow, feedwater flow, and generator output will reflect the steady state oscillation pattern. All appropriate annunciators will actuate if setpoint is reached.

Removal of the malfunction will restore the selected pressure regulator to normal operation.

REFERENCES: GEK 5585

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Electronic Pressure Regulator Failure

NUMBER: TC05

TYPE: Variable (0-100% Stroke)

CAUSE: Main Steam Pressure Transducer DT-4 Failure

PLANT CONDITIONS:

Reactor Startup or Power Operating Condition

EFFECTS: The output stroke of the electronic pressure regulator will go to the instructor specified value. If the stroke fails low, the control system will respond by closing the control valves. The mechanical pressure regulator will take over to limit the transient. The EPR servo will drive to zero.

If the instructor specified value of EPR stroke is failed high, the control valves will open to the mechanical 'Reactor Flow Limit' stop. Reactor pressure will decrease.

Removal of the malfunction will restore normal operation of the pressure transducer.

REFERENCES: GEK 5585 GE Turbine Manual
CWD B-191301, Sh. 110

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Mechanical Pressure Regulator Failure

NUMBER: TC06

TYPE: Variable (0-100% of Sensed Pressure Range)

CAUSE: Failure of the Mechanical Pressure Sensor

PLANT CONDITIONS:

Reactor Startup or Power Operation

EFFECTS: The mechanical pressure regulator sensed pressure input will go too the instructor specified value. If the MPR sensed pressure signal is failed low the MPR output stroke will increase, if the MPR stroke becomes greater than the EPR stroke it will cause the control valves to open, reactor pressure will decrease and the EPR will drive low.

Removal of malfunction TC06 will restore normal operation of the MPR.

REFERENCES: GEK 5585 GE Turbine Manual
CWD B-191301, Sh. 110

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Speed Control Unit Fails (High/Low)

NUMBER: TC07

TYPE: Discrete; A=High, B=Low

CAUSE: Speed Load Changer Fails to Maximum/Minimum

PLANT CONDITIONS:

Plant Startup

EFFECTS: This malfunction will result in the turbine speed control unit failing high/low due to failure of the speed/load changer. The speed/load changer controls the turbine speed when the generator is not connected to the bus, and can be used to load or unload the generator when it is supplying power to the system. The speed/load changer is equipped for motor or hand operation and is provided with two stops to limit its operating range within predetermined limits. If the speed/load changer fails high, the speed of the turbine-generator will increase to 107% speed at no-load with rated steam conditions. If the speed/load changer fails low, then the turbine-generator speed will be limited to 95% speed at no-load with rated steam conditions. However, if on the grid, failing high will result in additional loading of turbine-generator (provided MHC is not controlling turbine load) and failing low will result in unloading the turbine-generator. (MHC will open bypass valves to maintain reactor pressure). In the event the reactor is below rated conditions, a greater control valve opening is required to drive the turbine. All applicable alarms will actuate as conditions require.

Removal of the malfunction will return the speed control unit to normal.

REFERENCES: GEK 5585, GE Turbine Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Control Valve (1,2,3,4) Fails As-Is

NUMBER: TC08

TYPE: Generic (1,2,3,4)

CAUSE: Control System Failure

PLANT CONDITIONS:

Plant Startup/Shutdown or Power Operation

EFFECTS: This malfunction will cause the selected main turbine control valve(s) to fail as-is due to control system failure. The plant will respond to the event. The MHC System will compensate for the affected control valve(s) even to the point of bypass valve operation. The turbine-generator load will respond correctly to the event as evident by load conditions.

Removal of the malfunction will restore the control valve(s) affected to normal.

REFERENCES: GEK 5585, GE Turbine Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Stop Valve (1,2,3,4) Fails (Open/Closed)

NUMBER: TC09

TYPE: Generic (A,C,E,G) - FAILS OPEN
(B,D,F,H) - FAILS CLOSED

CAUSE: Control System Failure in Selected Stop Valve (Open/Closed)

PLANT CONDITIONS:

Plant Startup/Shutdown or Power Operation

EFFECTS: This malfunction will cause the selected stop valve to fail open(closed). The selected stop valve position will fail to the desired position if valve operating oil pressure is present. The appropriate valve indication will respond and also any perturbations which follow as a result of the event. Turbine stop valve closure with the reactor at power will result in significant pressure and reactivity transients. To limit the transients, a scram will be initiated when 3 or more stop valves are less than 90% open. The scram logic is arranged so that any one stop valve closure will not result in a 1/2 scram.

Removal of the malfunction will restore the affected stop valve to its original position.

REFERENCES: GEK 5585, GE Turbine Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Auxiliary Oil Pump Trip

NUMBER: TU01

TYPE: Discrete

CAUSE: Activation of Overload Device

PLANT CONDITIONS:

Plant Startup/Shutdown

EFFECTS: If running, or when started, the instantaneous overcurrent device will trip, the auxiliary oil pump motor will be deenergized and the pump tripped annunciator will actuate. If the turbine is running above about 1620 RPM, no other effects will be seen. If below about 1620 RPM oil header pressure will decrease causing the turning gear oil pump to start automatically. Bearing oil and hydraulic oil pressure indicators on CRP 9-7 will reflect aux. oil pump trip. If below 1620 RPM a loss of the auxiliary oil pump will result in a turbine trip with bypass valve failure.

Removal of the malfunction will restore normal operation of the pump and reset the overload device.

REFERENCES: CWD B-191301, Sh. 155
GEK 5585, GE Turbine Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Bearing (1-10) Temperature High

NUMBER: TU02

TYPE: Generic (1-10), Variable (100%=Normal Plus 150°F)

CAUSE: Lube Oil Flow Blockage

PLANT CONDITIONS:

Main Turbine Operating

EFFECTS: The reduced flow will cause the bearing temperature to increase at the rate of 10°F/min. when bearing oil temperature reaches 145°F and bearing metal temperature reaches 225°F their associated annunciators will actuate. As the oil temperature increases, its lubricating properties will decrease causing the heat generated by friction in the bearing to increase. Bearing metal temperature will increase faster than oil drain temperature.

If bearing metal temperature exceeds about 300°F, the bearing will wipe. Indicated temperature will spike suddenly as the babbit melts. The increased flow area through the bearing will allow increased oil flow and the loss of friction will allow bearing metal temperature to decrease rapidly. The temperature spike may not be seen on the multi-point recorder due to the long time between monitoring individual points.

The load which was carried by this bearing will be picked up by the adjacent bearings which will show a noticeable temperature increase. Turbine vibration will also show a significant increase as the affected bearing wipes. The adjacent bearing vibration will also show noticeable increases.

Removal of the malfunction will restore normal oil flow to the bearing. If the bearing has wiped, the Simulator must be reinitialized to remove the resulting effects.

<u>REFERENCES</u>: GEK 5585, GE Turbine Manual

VERMONT YANKEE NUCLEAR POWER CORPORATION

SIMULATOR MALFUNCTION

CAUSE AND EFFECTS

TITLE: Turbine Bearing (1-10) High Vibration

NUMBER: TU03

TYPE: Generic (1-10), Variable (100%=15 mils)

CAUSE: Rotor Unbalance

PLANT CONDITIONS:

Main Turbine Operating

EFFECTS: This malfunction will cause the selected shaft bearing vibration to increase by the instructor specified amount. The vibration levels of adjacent bearings will also increase. The increased vibration will be in addition to the normal or critical speed values. If vibration is allowed to increase, the turbine will trip due to high vibration. All associated alarms will actuate when appropriate. When the malfunction becomes active the selected bearing(s) vibration will increase at .5 mils/min. up to the severity chosen.

Removal of the malfunction will allow selected bearing vibration to decrease at .5 mils/min. down to their normal values.

REFERENCES: GEK 5585, GE Turbine Manual