REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards responses to remaining Reactor Sys Branch Questions SEE RPT
211.129,211.031,211.148 & 211.209.Revised FSAR pages re
ODYN code will be incorporated into Amend 23.

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January 11, 1982
G02-82-26
SS-L-02-CDT-82-008

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO REACTOR SYSTEMS BRANCH QUESTIONS
AND ODYN ANALYSIS WNP-2 FSAR REWRITE

Reference: Letter, R. L. Tedesco to R. L. Ferguson, "WNP-2 FSAR - Request for Additional Information", dated June 8, 1981.

Enclosed are sixty (60) copies of responses to the remaining Reactor Systems Branch questions. The responses to Questions 211.129 and 211.136 are new. The responses to 211.031 and 211.148 are rewrites of responses previously submitted to the NRC. The response to 211.209 was also submitted as a part of the LRG appendix (RSB-3). These responses will be incorporated into the FSAR in Amendment 23.

Also, enclosed are sixty (60) copies of the draft revised FSAR pages as a result of the ODYN analysis. These revised pages will also be incorporated into Amendment 23 to the WNP-2 FSAR.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/ct
Enclosures

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

13001
S 1/60

Q. 211.031
(5.4.7)

In Table 5.4-3 of the FSAR, you indicate that the RHR isolation valves MOF008 and MOF009 are closed upon generation of a signal indicating reactor low water level. It appears that you have mislabelled these valves in this table as "recirculation line suction" rather than as "RHR isolation". Indicate whether this valve isolation signal is based on the same signal as the RHR pump actuation in the low pressure coolant injection system (LPCI) mode (i.e., a water level which is 1.0 foot above the active core). If not, indicate the water level in the reactor pressure vessel at which the isolation signal is generated, thereby isolating the RHR suction valves. Show that cooling of the reactor core can be maintained assuming a pipe break outside the containment. Assuming a pipe break outside containment in the RHR system when the plant is in a shutdown cooling mode, provide the following additional information:

a. Identify the systems available for maintaining core cooling.

b. Indicate the maximum discharge rate resulting from the postulated break and the time interval available for recovery based on the discharge rate and its effect on core cooling.

c. Identify the alarms available to alert the operator in the event of such a break and show that sufficient time is available for operator action to prevent damage to safety-related systems.

d. Indicate what recovery procedures are available.

e. Following a postulated break in a moderate energy line, the single failure criterion should be applied in the manner discussed in Section 3.6.1 of the Standard Review Plan (SRP) NUREG-75/087, and in Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment", November 24, 1975.

Response:

~~a. F008 and F009 isolate at reactor water level 3 which is 180 inches above the top of the active fuel. LPCI is initiated at reactor water level 1 (Reference FSAR Figure 5.2-6.) Should a pipe~~

Insert Attached

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Insert to Page 211.031-1:

Table 5.4-3 has been revised to indicate that the RHR isolation valves MOF008 and MOF009 are the "Shutdown Cooling Suction" valves.

F008 and F009 isolate at reactor water level 3 which is 174 inches above the top of the active fuel. LPCI is initiated at reactor water level 1 (Reference FSAR Figure 5.2-6).

The following items respond directly to the items requested above:

a. should a pipe

failure occur, outside the containment, in the RHR system when the plant is in shutdown cooling, acceptable core cooling would be achieved by the core cooling systems. The following core cooling systems would be available to maintain core cooling when applying SRP 3.6.1 and BTP APCSB 3-1:

- If the single active failure is HPCS the following are available: LPCS + 2 LPCI + ADS.
- If the single active failure is LPCS the following are available: HPCS + 2 LPCI + ADS.
- If the single active failure is LPCI (not shutdown cooling loop) the following are available: HPCS + LPCS + 1 LPCI + ADS.

Insert attached

b. The maximum discharge resulting from the largest crack in the RHR piping outside containment is determined using the guidelines in BTP MEB 3-1 for moderate energy piping. The maximum discharge rate is estimated to be 1000 gpm (to be confirmed later in the ongoing pipe break and missile study). This is based upon a pipe break in the pump discharge piping (18" Schedule 30) at the pump discharge flange, normal water level in the reactor during shutdown cooling (approximately 50 inches below the steam line nozzles), reactor pressure of 135 psig, and the RHR pump running at 7450 gpm (normal shutdown flow). See "c" below for the time interval available for recovery. The flow rate used in the LaSalle analysis referenced in part a) of this response was 1443 gpm.

c. The following alarms are available to the operator in the event of a pipe break in the shutdown cooling line outside containment.

1. Low reactor water level alarm (Level 4, 198.7 inches above active fuel).
2. Low reactor water level (Level 3) to scram and isolate MOF008 and MOF009.
3. Equipment area high temperature (Class 1E) to isolate MOF008 and MOF009.
4. High flow rate in the shutdown cooling suction line to isolate MOF008 and MOF009 (Class 1E).

Insert to Page 211.031-2:

A special analysis for LaSalle was made of a hypothesized crack in the RHR suction line outside of primary containment during operation in the shutdown cooling mode. This analysis was performed with the standard GE LOCA models. For this event the realistic or actual system conditions are as follows.

No high pressure systems are available for water inventory restoration, i.e., no feedwater, no CRD flow, no HPCS, and no RCIC, but the reactor water level is at normal elevation at the start of this event. Vessel pressure is less than 150 psia and the MSIVs are closed at the start of this event. The decay heat is approximately 1% of rated power, i.e., approximately 4 hours have elapsed subsequent to reactor scram or shutdown.

For a conservative solution to this hypothetical event, the following sequence of events and conditions were assumed to exist or ensue from the hypothesized crack in the suction line:

a. Crack occurs in the RHR line; water level decreases to reactor vessel level 3; then the RHR isolation commences and is completed 40 seconds later.

b. System pressure rises as a result of the isolation to where the vessel pressure reaches the SRV setpoint thus causing them to open, blowdown, and reclose.

c. Inventory depletion results from blowdown and from leakage out of the cracked line.

d. The operator manually actuates ADS to reduce vessel pressure to where the low pressure ECCSs can replenish the water inventory.

e. Water level is restored to within normal limits to protect the core from over temperature.

Results are presented in Figures 211.031-1 through 211.031-4 for a bounding calculation of this event. The standard Appendix K assumptions were used along with these conservative initial conditions:

a. The timing index was started at the RHR isolation (when Level 3 was attained) to neglect the time for the level to fall from normal water level to level 3 (about 2 minutes).

b. An initial pressure of 1055 psia was assumed to neglect the pressure rise time from the 150 psia (pressure permissive for shutdown cooling) upon completion of RHR isolation to the 1055 pressure attainment. This results in increased mass loss during the 40 second isolation period due to greater driving pressure. It also decreases the time increment needed for pressure to attain the relief valve setpoint.

Insert to Page 211.031-2 (Continued):

c. The analysis assumes that scram occurs coincident with the start of the timing instead of 4 hours earlier. This assumption maximizes the peak clad temperature and steam production during the transient thus driving more fluid from the vessel and prolonging the blowdown phase.

d. Only one LPCS and one LPCI loop were assumed to be available throughout the event. Operator action does not include possible diversion of the other two LPCI loops from the RHR mode.

e. The crack area used in the analysis is defined consistently with the MEB 3-1 guidance for crack size. This crack area is consistent with FSAR postulates.

Results from this conservative analysis who that more than 20 minutes are available for the operator to depressurize the vessel. Once the system pressure is below the LPCI or LPCS shutoff head, the reactor water level is restored to normal limits very rapidly. The maximum clad temperature is much less than the arbitrary 2200°F limitation.

b. The RHR system is a low pressure system, and all of the piping outside of the primary coolant pressure boundary is classified as "moderate energy" piping and, according to BTP MEB 3-1, only cracks (i.e., not breaks) are considered in moderate energy piping. Reactor vessel pressure must be decreased to below 135 psig before the RHR system can be connected to the reactor vessel.

5. Reactor building floor drain sump level for leakage rates greater than 50 gpm.

6. Reactor building floor drain leakage rate alarm (5 gpm).

7. ECCS pump room flood level instrumentation (Class 1E), installed to detect passive failures in the ECCS post-LOCA (Reference response to FSAR Question 212.003).

Notwithstanding these alarms, however, only about 13,000 gallons of water will spill out of the break before the reactor water level drops from the normal shutdown cooling level to Level 3, automatically closing MOF008 and MOF009 and isolating the break. No single active failure can prevent isolation of the break. ~~During the time of the pipe break, HPCS and RCIC will not automatically initiate, because their initiation signal (Level 2) is about 50 inches below Level 3. Also, RCIC will not automatically initiate because the reactor pressure (135 psig maximum during shutdown cooling) is too low to operate the RCIC turbine~~.

For the largest pipe break (1000 gpm), which can only occur in the RHR A or B pump rooms, the flood level resulting from 13,000 gallons will not affect operation of either RHR pump A or B. In addition, the flooding would only affect the room in which the pipe break occurred because of the watertight integrity of the RHR A and B pump rooms. Therefore, no operator action is required to protect these pumps.

Pipe breaks in the RHR shutdown cooling mode which can affect other ECCS systems (LPCS, HPCS or RHRC) via flooding through the floor drain piping in the upper portions of the reactor building will have flooding rates less than 1000 gpm because they will be higher in the building (less static head), have less driving head due to friction losses, and smaller crack sizes (smaller diameter pipe). These pipe breaks in the upper portion of the reactor building will be immediately detected by the high flow (5 gpm) alarms in the floor drain downcomer piping. Regardless of what the pipe break discharge rate is, the flood level resulting from 13,000 gallons is not capable of affecting the operation of the LPCS,

HPCS or RHRC pumps, assuming all of the water is spilled into each pump room. Again, no operator action is required to protect these pumps.

It should be noted that the environmental effects (pressure, temperature and humidity) of pipe breaks during shutdown cooling are being addressed by the ongoing pipe break and missile study.

Insert attached

~~d. Core cooling is not a concern. Sufficient ECCS~~ equipment remains functional to automatically keep the core covered at all times. The cold shutdown procedure, i.e., containment heat removal, needs to be resumed. If the pipe break disables the common shutdown cooldown suction line, cold shutdown can be assumed by the alternate shutdown cooldown path discussed in 15.2.9. If the pipe break in the shutdown cooldown line is downstream of the F006 valve, normal shutdown cooldown can be resumed using the redundant RHR ~~shutdown loop.~~

e. For application of single failure criteria, see "a" above.

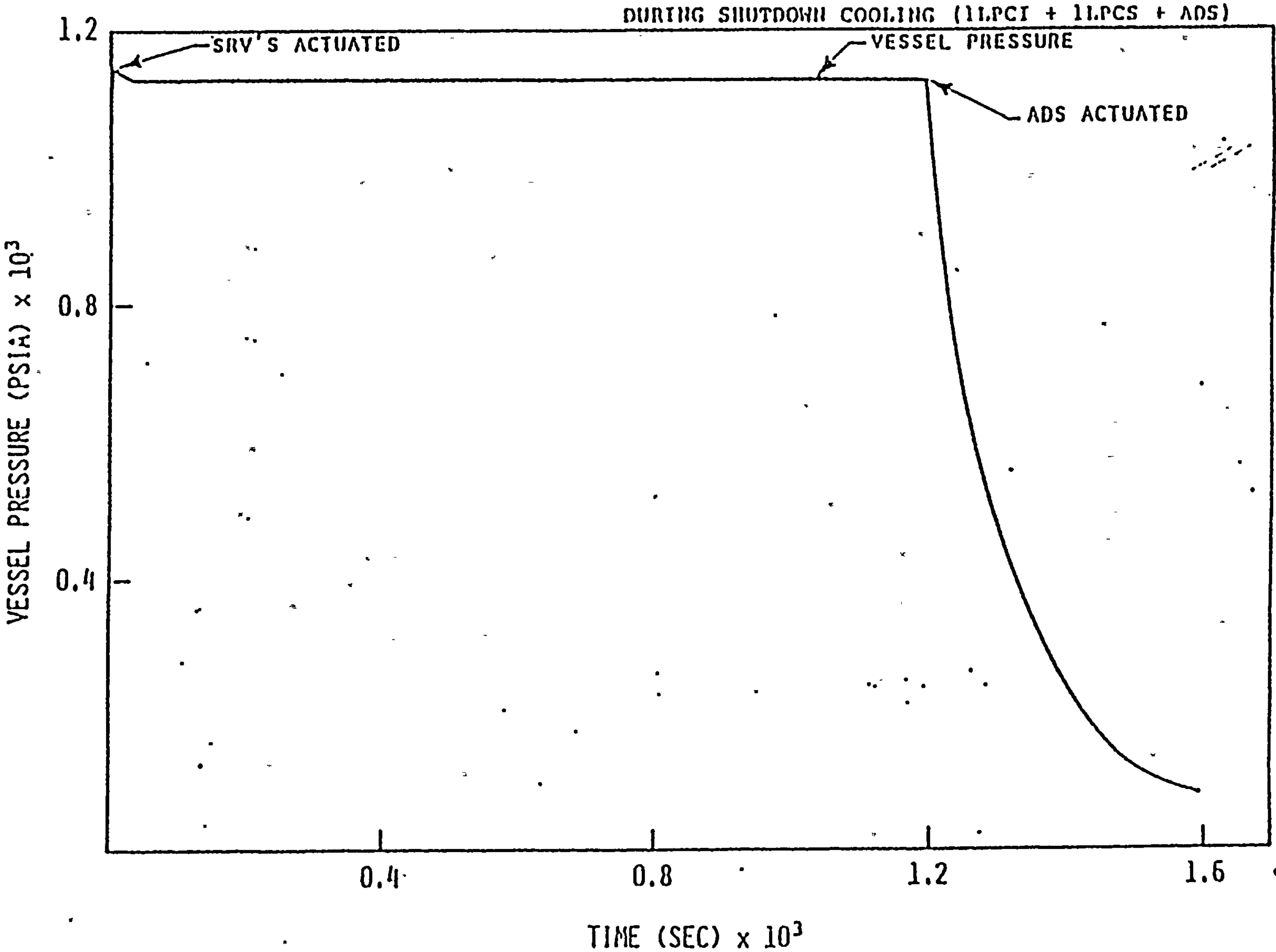

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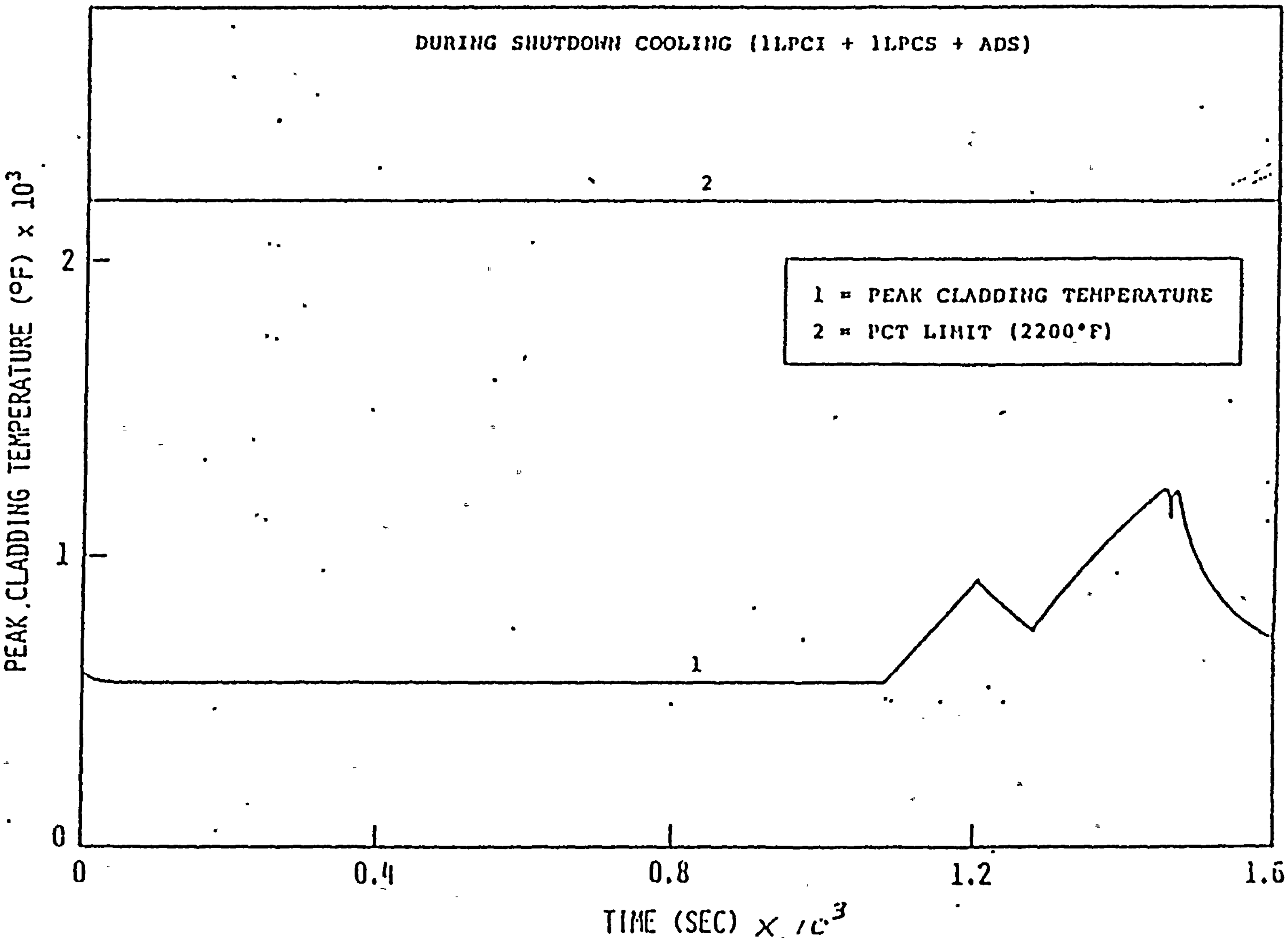
FIGURE Q211.031-1
VESSEL PRESSURE VS. TIME FOR A CRACK IN THE RHR LINE

Insert to Page 211.031-4:

d. If a break should occur in one RHR shutdown cooling loop outside containment during shutdown, the following action is taken upon detection and isolation. The main steam isolation valves will be reopened and reactor excess steam will blow down to the main condenser until the shutdown cooling process via the other RHR loop is established.

The redundant shutdown cooling loop components are also not assumed to fail under the cited NRC requirements of BTP APCSP 3-1.

If the pipe crack should occur in the common manifold supplying both redundant loops, the isolation mechanism is the same as before, but recovery would require reversion to the alternate shutdown configuration discussed in 15.2.9. In this configuration, vessel water is circulated from the suppression pool through the RHR heat exchanger to the vessel with return to the suppression pool via the ADS discharge lines.



Q212.133-6

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FIGURE [illegible]-3
Q 211.031
PEAK CLADDING TEMPERATURE VS.
TIME FOR A CRACK IN THE RHR LINE

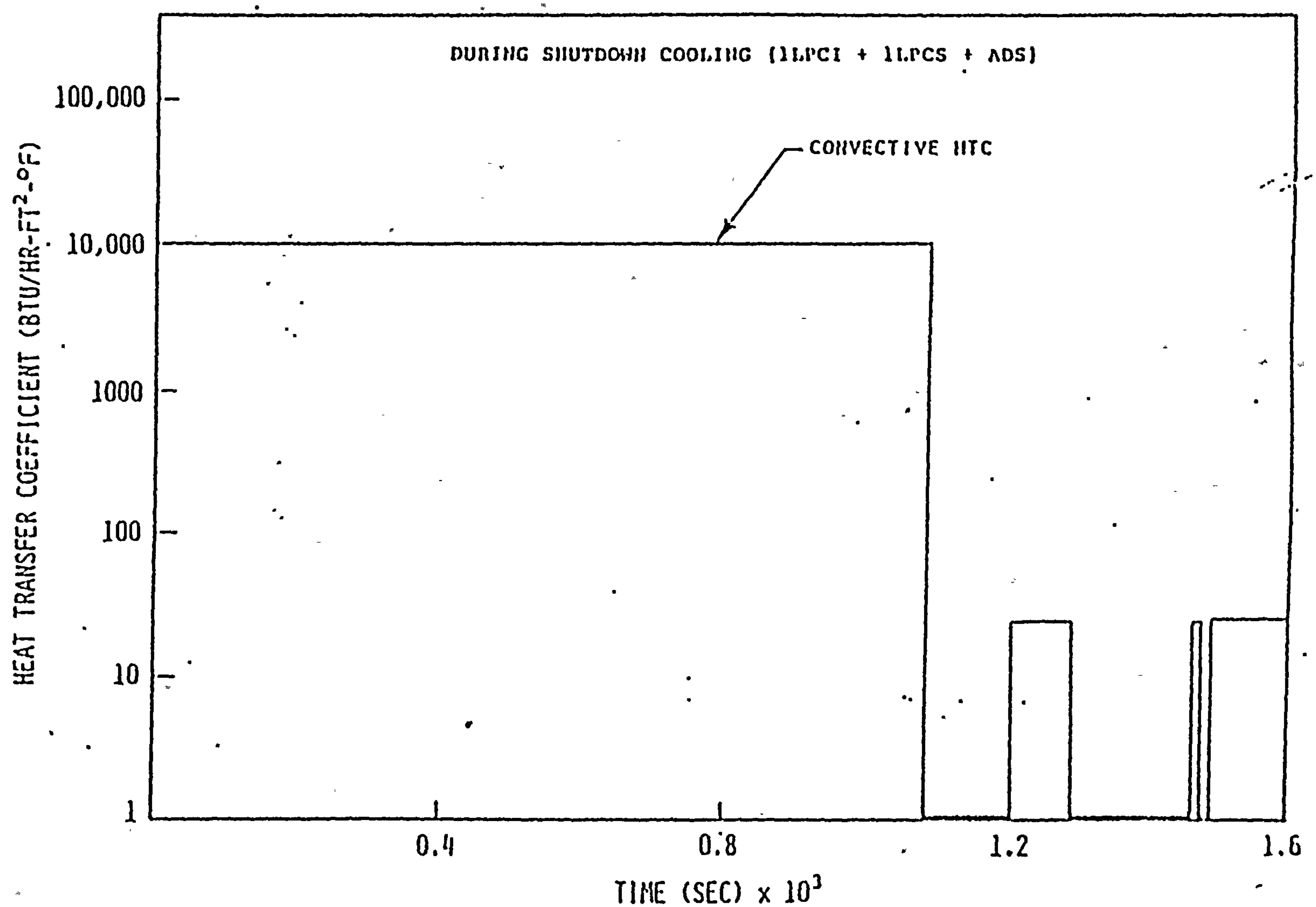

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FIGURE Q211.031-4
HTC AT PCT NODE VS. TIME FOR A CRACK IN THE RHR LINE

Q. 211.129
(4.6)

The standby liquid control system and the recirculation flow control system are reactivity control systems. Address or reference these systems in Section 4.6 and address all requirements of Standard Review Plan 4.6.

Response:

The recirculation flow control system is evaluated against the general design criteria as follows:

a. Criteria 20, 21, 23 and 25: Criteria 20, 21, 23 and 25 are applicable to protection systems only. The recirculation flow control system is a reactivity control system but is not a protection system.

b. Criterion 26: The recirculation flow control system is the second reactivity control system required by this criterion. The requirements of this criterion do not apply within the system itself.

c. Criterion 27: The recirculation flow control system is not intended to control reactivity following an accident. Consequently, this criterion does not apply.

d. Criterion 28: The transient analyses in Chapter 15 evaluate the consequences of reactivity events involving changes in reactor coolant temperature and pressure and cold water addition. The results of these analyses indicate that none of these postulated events causes damage to the reactor coolant pressure boundary. In addition, the integrity of the core, its support structures and other reactor pressure vessel internals are maintained so that the capability to cool the core is assured.

The evaluations with respect to general design criteria of the standby liquid control system can be found in Section 9.3.5.*

The first paragraph of Section 4.6 has been replaced with the following:*

"Functional design of the control rod drive system (CRD) is discussed below. Functional designs of the recirculation flow control system and standby liquid control system are described in Sections 5.4.1 and 9.3.5, respectively.

*Revised draft FSAR pages attached

Revise for BRSCN 81-381

TABLE 5.4-3

RCPB PUMP AND VALVE DESCRIPTION*

Location	Active/ Inactive	Valve No.	Reference Figure
Valve Description			
RHR Vessel In	Active	E12F041	5.4-13a and b
	Active	E12F042	5.4-13a and b
	Inactive	E12F111	5.4-13a and b
RHR/Recir-culation Line In	Active	E12F050	5.4-13a and b
		E12F099	5.4-13a and b
	Active	E12F053	5.4-13a and b
	Inactive	E12F112	5.4-13a and b
Head Spray	Active	E12F019	5.4-13a and b
	Active	E12F023	5.4-13a and b
RHR SHUTDOWN COOLING ~~[illegible]~~	Active	E12F009	5.4-13a and b
~~[illegible]~~	Active	E12F008	5.4-13a and b
Suction	Inactive	E12F113	5.4-13a and b
RCIC Vessel Out	Active	E51F064	5.4-9a and b
	Active	E51F063	5.4-9a and b
	Active	E51F076	5.4-9a and b
	Active	E51F008	5.4-9a and b
(Nuclear Boiler) Reactor Vessel Head	Inactive	B22F001	5.1-3c
	Inactive	B22F002	5.1-3c
	Inactive	B22F005	5.1-3c
Feedwater in	Active	B22F010	5.1-3c
	Inactive	B22F011	5.1-3c
	Active	B22F032	5.1-3c
	Active	B22F065	5.1-3c
Safety Relief	Active	B22F013	5.1-3c

*In addition to the process valves listed herein, there are instrument test connections, drain valves and sampling valves less than one inch nominal size within the RCPB. Refer to figures in 3.2 for their locations.

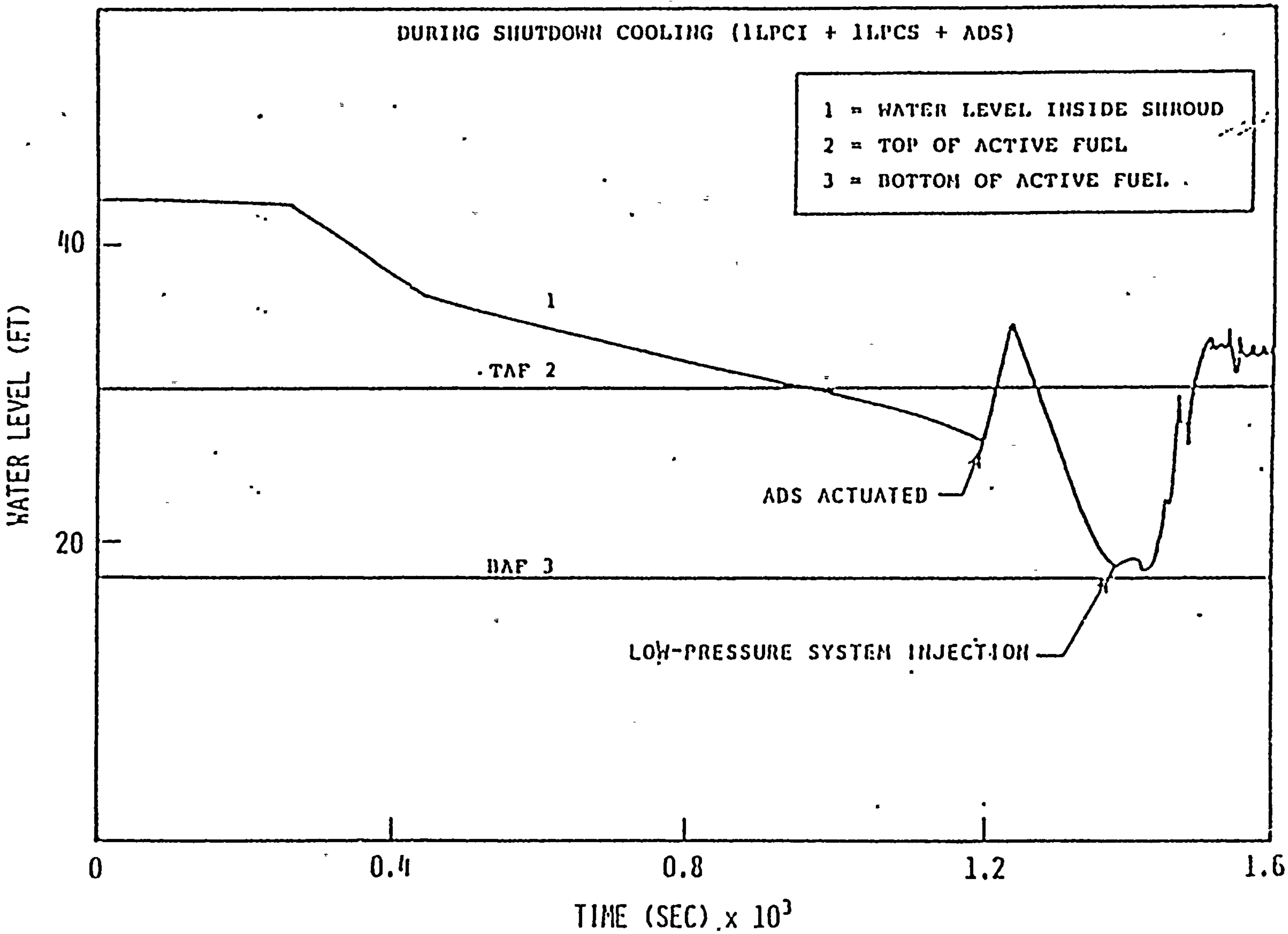

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FIGURE Q211.031-2
WATER LEVEL VS. TIME FOR A CRACK IN THE RHR LINE

Insert attached

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

~~The reactivity control systems consist of control rods and control rod drives, supplementary reactivity control for the initial core, and the standby liquid control system. The SLC system is described in 9.3.5.~~

4.6.1 INFORMATION FOR THE CRD SYSTEM

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

4.6.1.1.1.1 General Design Bases

4.6.1.1.1.1.1 Safety Design Bases

The control rod drive mechanical system meets the following safety design bases:

a. The design provides for a sufficiently rapid control rod insertion that no fuel damage results from any abnormal operating transient.

b. The design includes positioning devices, each of which individually supports and positions a control rod.

c. Each positioning device:

1. Prevents its control rod from initiating withdrawal as a result of a single malfunction.

2. Is individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

3. Is individually energized when rapid control rod insertion (scram) is signaled so that failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

4.6.1.1.1.1.2 Power Generation Design Basis

The control rod system drive design provides for positioning the control rods to control power generation in the core.

Insert

Functional design of the control rod drive system (CRD) is discussed below. Functional designs of the recirculation flow control system and standby liquid control system are described in Sections 5.4.1 and 9.3.5, respectively.

Q. 211.136
(4.6.2)

Identify the specific common mode failure analysis and protection from common mode failures referenced in Section 15A by Sections 4.6.2.1 and 4.6.2.2, respectively.

Response:

Section 15A is the Plant Nuclear Safety Operational Analysis (NSOA). This analysis provides analytically determinable limits on the consequences of different classifications of plant events i.e., expected operational transients; unexpected operational transients, and is thus an event-consequence oriented evaluation.

Event 53 - Reactor Shutdown and Cooldown Without Control Rods satisfies the requirements of Regulatory Guide 1.70 for Sections 4.6.2.1 and 4.6.2.2 respectively. In Section 15A, this event is discussed on page 15.A.6-36.

In addition, the scram discharge volume system has been evaluated (as requested in Question 010.041) against the criteria enumerated in the Generic Safety Evaluation report "BWR Scram Discharge System", dated December 1, 1980. With incorporation of the system modification described in the response to question 010.041, the scram discharge volume system was found to be in full compliance with the SER.

component in the SLC system and the fact that nuclear system cooldown takes several hours while liquid control solution injection takes approximately two hours. Since this probability is small, considerable time is available for repairing and restoring the SLC system to an operable condition while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by demonstrating operation of the operable pump.

The SLC system is evaluated against the applicable General Design Criteria as follows:

Criterion 2:

The SLC system is located in the area outside of the primary containment and below the refueling floor. In this location, it is protected by walls from external natural phenomena such as earthquakes, tornadoes, hurricanes and floods and also from the effects of internal postulated accident events.

Criterion 4:

The SLC system is designed for the expected environment in the compartment in which it is located. In this compartment, it is not subject to the conditions postulated in this criterion such as missiles, whipping pipes, and discharging fluids.

Criteri~~on~~a ~~21~~ 20, 21, 23 and 25:

~~Criterion 21 is~~ These criteria are applicable to protection systems only. The SLC system is a back-up reactivity control system and is evaluated against Criteri~~on~~a ~~29~~ 26, 27, 28 and 29 (see below).

Criterion 26:

The SLC system is ~~the second~~ a backup reactivity control system ~~re-quired by this criterion~~ for the normal reactivity control systems. The requirements of this criterion do not apply to the SLC system.

Criterion 27:

This criterion is ~~not~~ applicable to the SLC system. See the General Design Criteria Section for discussion of combined capability.

Criterion 28:

This criterion is not applicable to the SLC system. ~~See the General Design Criteria Section for discussion of~~

Q. 211.148
(15.0)

Resolve the following items in Table 15.0-2:

a) Modify the values of vessel level trip to agree with the values specified in Figures 5.2-6 and 5.3-2 (item 29).

b) Specify the maximum percent relieving capacity assumed in Chapter 15 for each mode of SRV actuation (items 25 and 26).

c) Provide the following information concerning the high flux trip setpoint used as input to the REDY model (item 29):

   1) Explain why the high flux trip setpoint should not be increased to 122% NBR prior to multiplication by the thermal-power correction factor of 1.043 to account for the setpoint plus calibration error, instrument accuracy, and transient overshoot specified in Table 7.2-4.

   2) Explain why the thermal-power correction factor is applied to the high flux trip setpoint used in the REDY model.

d) Provide the following information concerning the APRM thermal trip setpoint used as input to the REDY model (item 30):

   1) Specify the highest flow-rated trip setpoint to be given in the Technical Specifications and how this value is obtained.

   2) Is the 122.03 NBR setpoint equal to the setpoint to be specified in step d (1) times the thermal power factor of 1.043 specified in step c (1)?

e) Table 15.0-2 does not contain all of the input parameters used in the REDY computer code. For each transient and accident analyzed in Chapter 15, provide the following:

   1) A list of all input parameters.

   2) Justification that the input parameters are conservative.

Response:

a) The water level setpoints specified in Figures 5.2-6 and 5.3-2 are consistent with those values specified in item 29 of Table 15.0-2. The apparent discrepancy is due to the different elevation level each table is referenced to.

Water Level Setpoints In	Reference To	Elevation Above Reactor Vessel Zero
Figure 5.2-6	Water level instrumentation zero	527.5 in.
Figure 5.3-2	Reactor vessel zero	0
Table 15.0-2	Bottom of steam separator skirt	514.0 in.

However, for the purpose of consistency and clarity, Table 15.0-2 has been revised to the same reference point as Figure 5.2-6 (bottom of the steam dryer skirt).

b) The maximum relieving capacity for each mode of SRV actuation assumed in Chapter 15 are:

Relief valve capacities	@1106 psig	is	101.8% NBR
Safety valve cpaacities	@1213 psig	is	111.5% NBR

These values have been added to item 22 of Table 15.0-2.

c) 1) The high flux setpoint shown in Table 7.2-4 is incorrect and a text correction is currently underway to make Table 7.2-4 consistent with Plant Technical Specifications. A correct list of setpoint specifications will be found in Table 2.2.1-1 of Chapter 16 when the WNP-2 Technical Specifications are completed. In the list, the trip setpoint column in Table 2.2.1-1 of Chapter 16 will correspond to the setpoint column in Table 7.2-4, and similarly the allowable values column to the setpoint + instrument drift column. The neutron flux (run model) shown in the new list is 120% of rated power that includes instrument drift. After accounting for calibration error and instrumentation inaccuracy, this setpoint totaled 121% of rated thermal power. Item 27 of Table 15.0-2 has been corrected as follows:

High Flux Trip % NBR Analysis Setpoint (121 x 1.043), % NBR = 126.20

A change in this high flux trip setpoint would cause no impact on transient results since for each transient analyzed in Chapter 15, the reactor was tripped by the direct scram prior to the high flux setpoint being reached.

2) The thermal power multiplier (RST) is used to give a conservative margin that is proportional to the core power.

d) 1) The maximum flow related trip setpoint given in Technical Specification will be 115.5% of rated thermal power (Table 2.2.1-1 of Chapter 16). This value is obtained by adding 2% instrument drift to the maximum nominal trip setpoint of 113.5%. The safety limit setpoint used in the FSAR analysis is 117%, which also includes instrument inaccuracy and calibration error.

2) Yes, the 122.03% NBR setpoint is calculated from multiplying the thermal power factor of 1.043 with the 117% safety limit as discussed in d)1) above.

e) 1) Table 15.0-2 was selected and provided to show the principal parameters related to the transient analysis. Providing a complete listing of inputs would be impractical. If some particular area of input is of special interest it can be provided upon specific request.

2) Parameters in which variation might have significant effect on the transient result were selected conservatively to bound the design values with uncertainty allowance.

The letter, R. H. Buchholz (GE) to P. S. Check (NRC), dated September 5, 1980, "Response to NRC Request for Information on ODYN Computer Model", list the input parameters of ODYN. These parameters coupled with Table 15.0-2 should enable the review for conservatism of REDY to be completed. REDY and ODYN have as input parameters much the same values. Qualification of the REDY computer code is documented in NEDO-10802.

TABLE 15.0-2 - (Continued)

		REDY	ODYN
17.	Void Coefficient (-)¢/% Rated Voids		
	Nominal EOC-1	7.48	7.48
	Analysis Data for Power Increase Events	12.70	**
	Analysis Data for Power Decrease Events	7.065	**
18.	Core Average Rated Void Fraction, %	41.32	41.37
19.	Scram Reactivity, $\$\Delta k$ Analysis Data	Figure 15.0-2	**
20.	Control Rod Drive Speed, Position versus time	Figure 15.0-2	Fig 15.0-2
21.	Jet Pump Ratio, M	2.41	2.41
22.	Safety/Relief Valve Capacity, % NBR		
	~~@ 1213 psig~~	~~111.5~~	
	Safety valve capacity	111.5	111.5
	Relief valve capacity	101.8	101.8
	Manufacturer	Crosby	Crosby
	Quantity Installed	18	18
23.	Relief Function Delay, seconds	0.4	0.4
24.	Relief Function Response, seconds	0.1	0.1
25.	Set Points for Safety/Relief Valves		
	Safety Function, psig	1177, 1187, 1197, 1207, 1217	1177, 1187, 1197, 1207, 1217
	Relief Function, psig	1091, 1101, 1111, 1121, 1131	1106, 1116, 1126, 1136, 1146
26.	Number of Valve Groupings Simulated		
	Safety Function, No.	5	5
	Relief Function, No.	5	5
27.	High Flux Trip, % NBR		
	Analysis set point (~~120~~ 121 x 1.043), % NBR	~~125.16~~ 126.2	126.2

TABLE 15.0-2 (Continued)

		REDY	ODYN
28.	High Pressure Scram Set Point, psig	1071	1071
29.	Vessel Level Trips, ~~Feet~~ Inches Above ~~Separator~~ Dryer Skirt Bottom		
	Level 8 - (L8), ~~feet~~	~~5.750~~ 55.5"	59.5"
	Level 4 - (L4), ~~feet~~	~~3.750~~ 31.5"	31.5"
	Level 3 - (L3), ~~feet~~	~~2.167~~ 12.5"	12.5"
	Level 2 - (L2), ~~feet~~	~~(-)2.041~~ (-)38"	(-)38"
30.	APRM Thermal Trip		
	Set Point, % NBR @ 100% core flow	122.030	122.030
31.	Recirculation Pump Trip Delay, Seconds	0.140	0.19
32.	Recirculation Pump Trip Inertia time constant for analysis, seconds*	6	6

* See Table 15.0-3.

** ODYN values are calculated within the code for end of cycle / condition.

*** The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_o n}{g T_o}$$, where t = inertia time constant (Sec).

J_o = pump motor inertia (lb-ft^2).

n = rated pump speed (rps).

g = gravitational constant (ft/sec^2).

T_o = pump shaft torque (lb-ft).

Q. 211.209
(5.2.2)
(6.3)

Provide assurance that your relief valve design is qualified (including testing after being subjected to an environment representative of an extended time period at normal operating conditions) to support your assumption that six of the seven ADS valves will operate. A quantitative history of safety/relief valve operation, including similar valves in other plants, should be included in this evaluation.

Response:

Presently valves of a similar but earlier design are installed and have been operated in Chinshan 1 and 2 and two SRVs of a modified design are in Browns Ferry 3. No unsatisfactory performance has been experienced on the modified design. A spare SRV of the earlier design, which was installed into Chinshan 1 and 2 was reported to have failed to fully reclose after a relief operation. The Chinshan 1 SRV did reclose with no further anomlies noted. A question exists as to whether gross leakage due to foreign material existed or if in fact the SRV did not fully reclose. A direct means of determining SRV position was not used. A Chinshan 2 SRV failed to reclose but did reclose fully after depressurizing the air inlet supply source. The failure to reclose has been attributed to a faulty solenoid and air valve assembly. The design of the SRVs to be installed into Hanford 2 are a modified version of those installed in Chinshan 1 and 2.

The response to Question 211.051 contains additional information related to the qualification of Crosby valves.

REVISED PAGES FOR ODYN ANALYSIS

TABLE 4.4-1 (continued) Page 2 of 2

General Operating Conditions

Average heat flux, Btu/hr-sq ft	145,100
Design operating minimum critical power ratio (MCPR)	1.24 (SEE TABLE 15.0-3)
Core inlet enthalpy at 420°F FFWT, Btu/lb	527.6
Core inlet temperature, at 420°F FFWT, °F	533
Core maximum exit voids within assemblies, %	76.0
Core average void fraction, active coolant	0.418
Maximum fuel temperature, °F	3,435.0
Active coolant flow area per assembly, in.^{2} (BOL)	15.824
Core average inlet velocity, ft/sec	6.88
Maximum inlet velocity, ft/sec	7.28
Total core pressure drop, psi	24.74
Core support plate pressure drop, psi	20.32
Average orifice pressure drop	
Central region, psi	6.03
Peripheral region, psi	16.54
Maximum channel pressure loading, psi	13.28

LIST OF FIGURES

b. The rated capacity of the pressure relieving devices are sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.1 x 1250 psig = 1375 psig) for the following events:

   (1) generator load rejection

   (2) turbine trip

   (3) loss of condenser vacuum

   (4) turbine trip without bypass - low power

   (5) closure of all main steam isolation valves

   (6) pressure regulator failure - fail open

   (7) loss of auxiliary power

   (8) feedwater controller failure - maximum flow

c. Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence from discharge piping losses.

Table 5.2-7 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics (either a point kinetics or ~~a~~ one dimensional kinetics simulation of the reactor core dynamics), the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These are represented with all their principal nonlinear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

A detailed description of ~~this~~ the models ~~is~~ are documented in licensing topical report NEDO-10802 (5.2.6, Ref. 5.2-1) and NEDO 24154. Safety/relief valves are simulated in the nonlinear representation,

and the models thereby ~~allows~~ allow full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

The typical capacity characteristic as modeled is represented in Figure 5.2-2 for the spring mode of operation. The associated turbine bypass, turbine control valve, and main steam isolation valve characteristics are also simulated in the models.

The associated bypass, turbine control valve, main steam isolation characteristics and ATWS pump trip are also represented fully in the models.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions.

5.2.2.2.2.1 Operating Conditions

a. operating power - 346~~3~~4 MWt (~~104.28~~ 104.25% of nuclear boiler rated power), — REDY

3468 MWt (104.3% of reactor rated power) — ODYN

b. vessel dome pressure - 1020 psig, and

c. steamflow - 14.98×10^6 lb/hr. (105% of nuclear boiler rated steam flow)

These conditions are the most severe because the maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steam isolation valves and a turbine-generator trip with a coincident ~~closure~~ Failure of the turbine steam bypass system valves, that represent the most severe abnormal operational transients resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams; therefore, it is used as the overpressure protection basis event and shown in Figure 5.2-1. Table 5.2-10 lists the sequence of events of the various systems assumed to operate during the main steam line isolation closure with flux scram event.

5.2.2.2.2.3 Scram

a. scram reactivity curve - ~~Figure 5.2-3~~ and

REDY: SEE Figure 5.2-3
ODYN: SEE Figure 5.2-1a

(time of scram is approximately 1.6 sec)

b. control rod drive scram motion - Figure 5.2-3

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specifications

a. Valve groups:

Spring-action safety mode - 5 groups

b. Pressure setpoint (maximum safety limit):

Spring-action safety mode - 1177 - 1217 psig

The setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically assumed setpoints in the analysis are 1 to 2% above the actual nominal setpoints. High conservative safety/relief valve response characteristics are also assumed.

5.2.2.2.2.5 Safety Valve Capacity

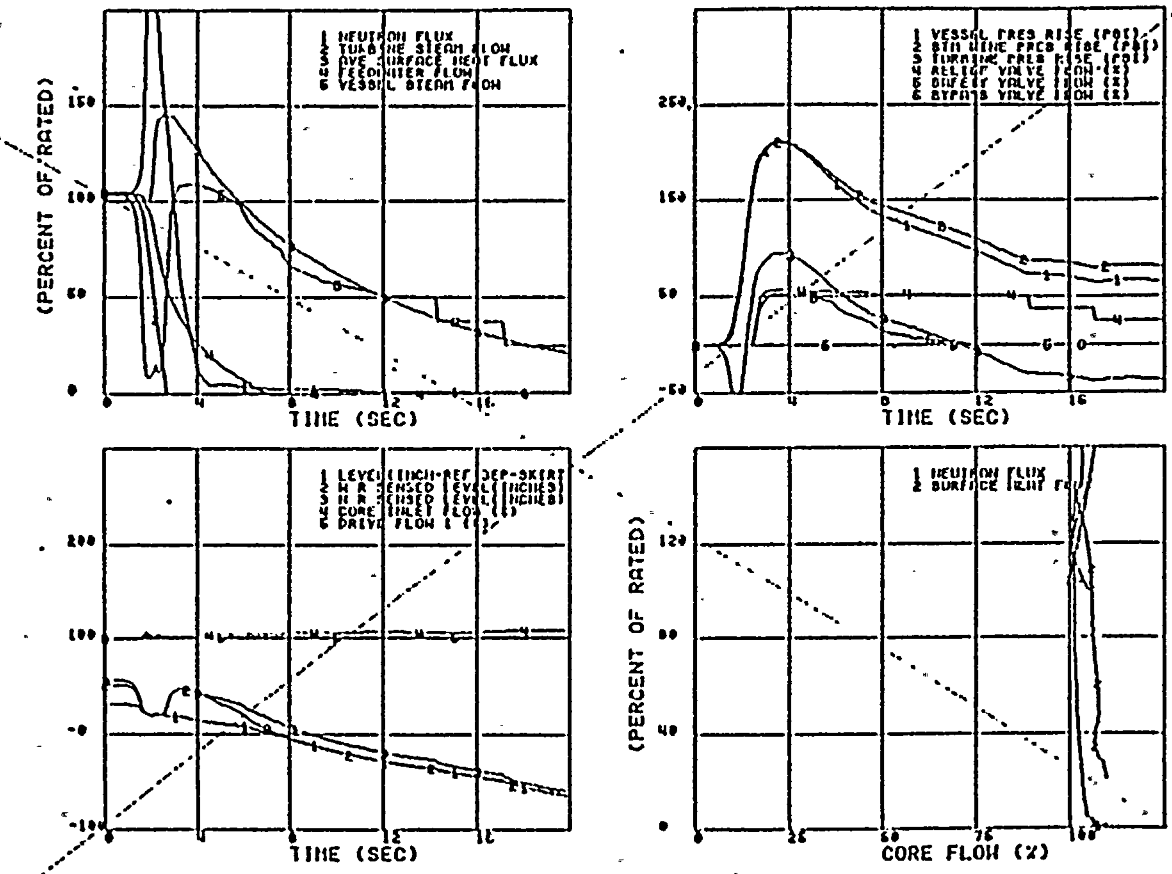
Sizing of the safety valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients in 5.2.2.2.2.2.

5.2.2.2.3 Evaluation of Restults

5.2.2.2.3.1 Safety Valve Capacity

The required safety/relief valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operting at the turbine-generator design conditions at a maximum vessel dome pressure of 1020 psig. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shutdown by the backup, indirect, high neutron flux scram. For the analysis, the spring-action safety setpoints are to be in the range of 1177 to 1217 psig. ~~The analysis~~ Both REDY and ODYN analyses indicate~~s~~ that the design valve capacity is capable of maintaining adequate margin below the peak ASME code allowable pressure in the nuclear system (1375 psig). Figure 5.2-~~1~~ 1 shows curves produced by ~~this~~ the REDY analysis, and Figure 5.2-1a shows the result of the ODYN analysis. The sequence of events in Table 5.2-10 and Table 5.2-10a assumed in these analyses were ~~assumed in this analysis was~~ investigated to meet code requirements and to evaluate the pressure relief system exclusively. As seen in the time response plot of the reactor vessel bottom head pressure in Figure 5.2-5, more pressure safety margin is predicted by the ODYN model, ~~which is more extensive,~~ than by the REDY model.

Under the General Requirements for Protection Against Overpressure as given in Section III of the ASME Boiler and Pressure Vessel Code, credit can be allowed for a scram from the reactor protection system. In addition, credit is also taken for the protective circuits which are indirectly derived when determining the required safety/relief valve capacity. The back-up reactor high neutron flux scram is conservatively applied as a design basis in determining the required capcity of the pressure relieving dual purpose safety/relief valves. Application of the direct position



REPLACE WITH NEW FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	OVERPRESSURE PROTECTION ANALYSIS (MSIV CLOSURE WITH HIGH FLUX SCRAM TRIP).	FIGURE 5.2-1

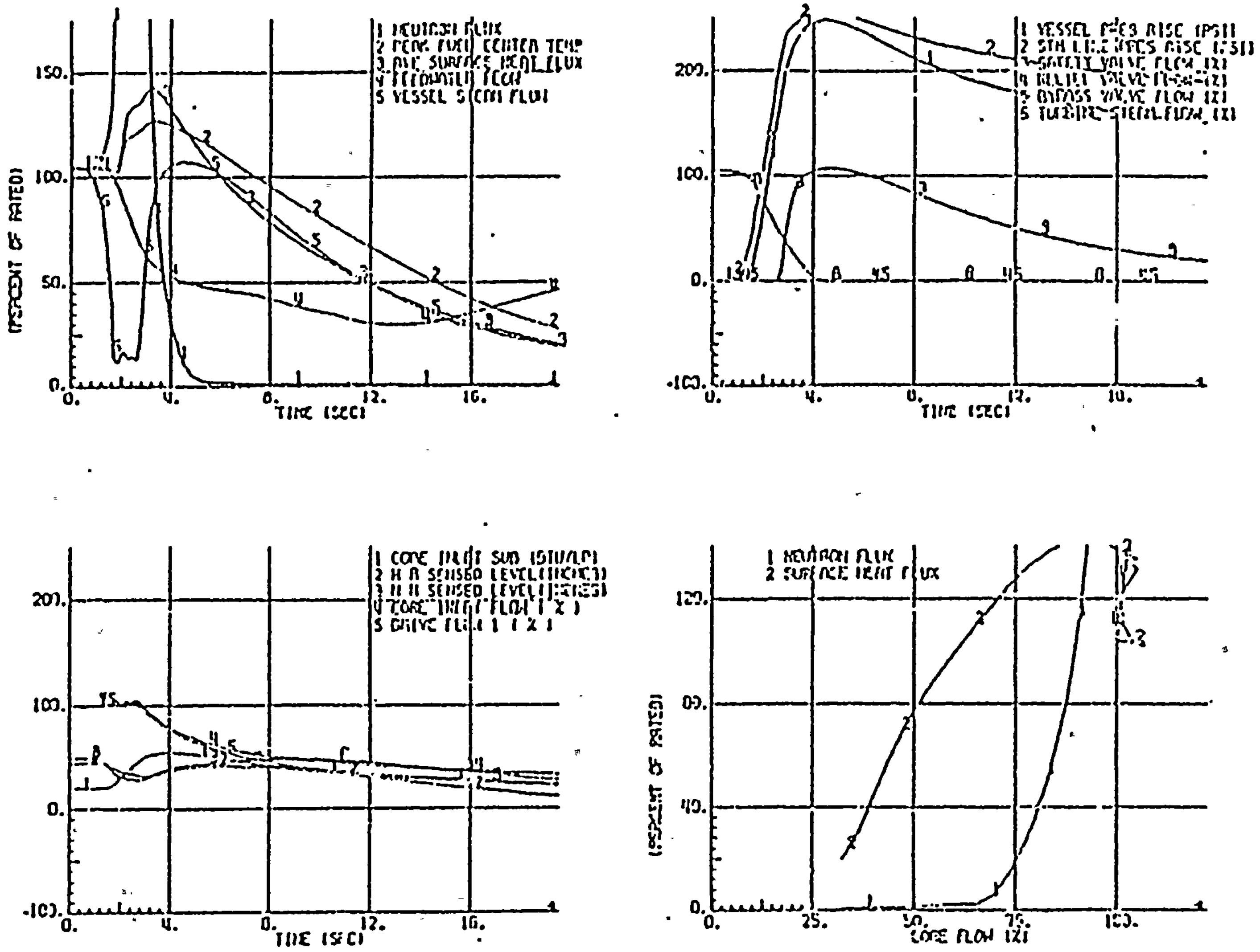


Figure 5.2-1 MSIV Closure with Flux Scram Installed Safety/Relief Valve Capacity
(REDY)

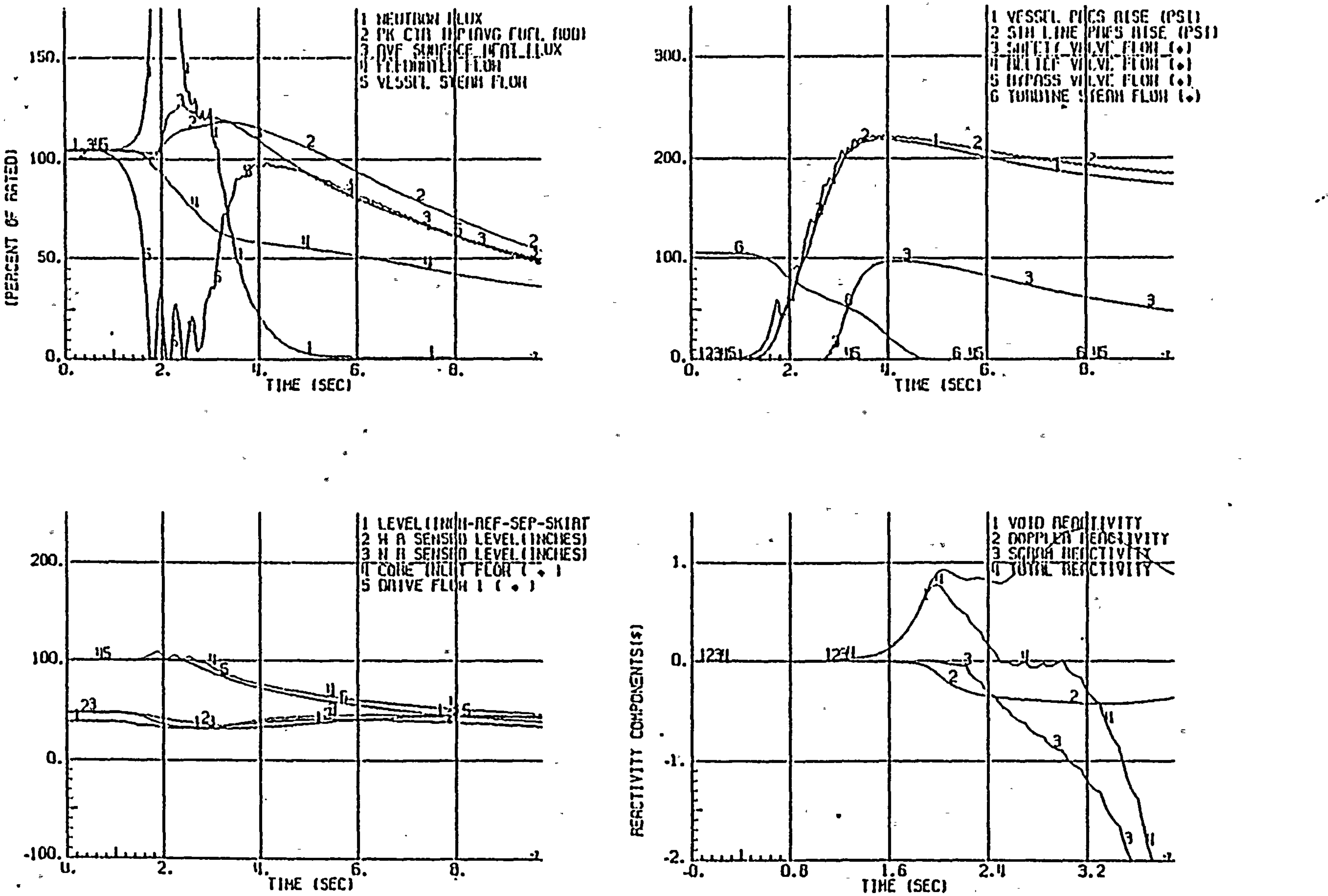


FIG. 5.2-1a

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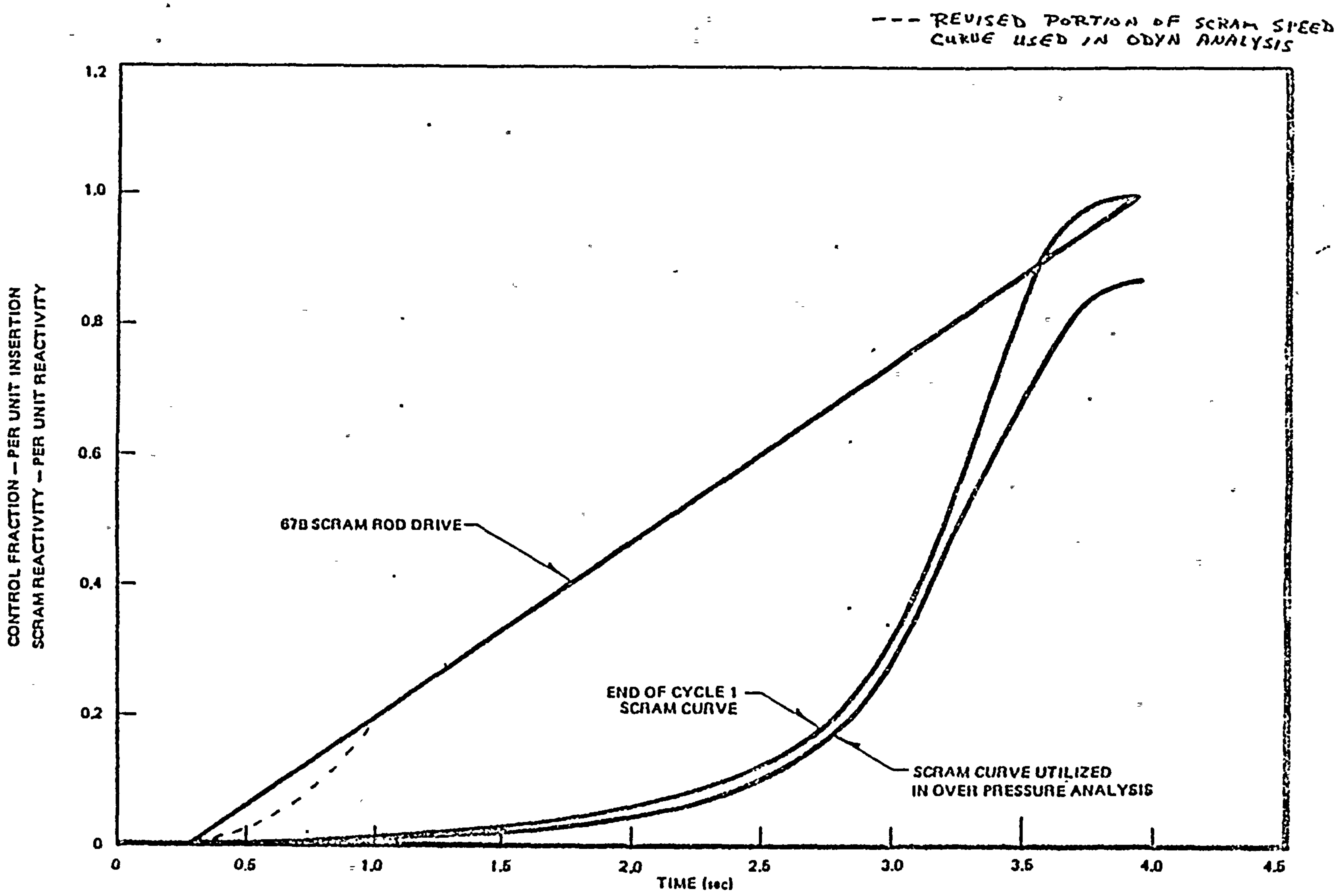
MSIV CLOSURE, HIGH FLUX SCRAM

(ODYN)

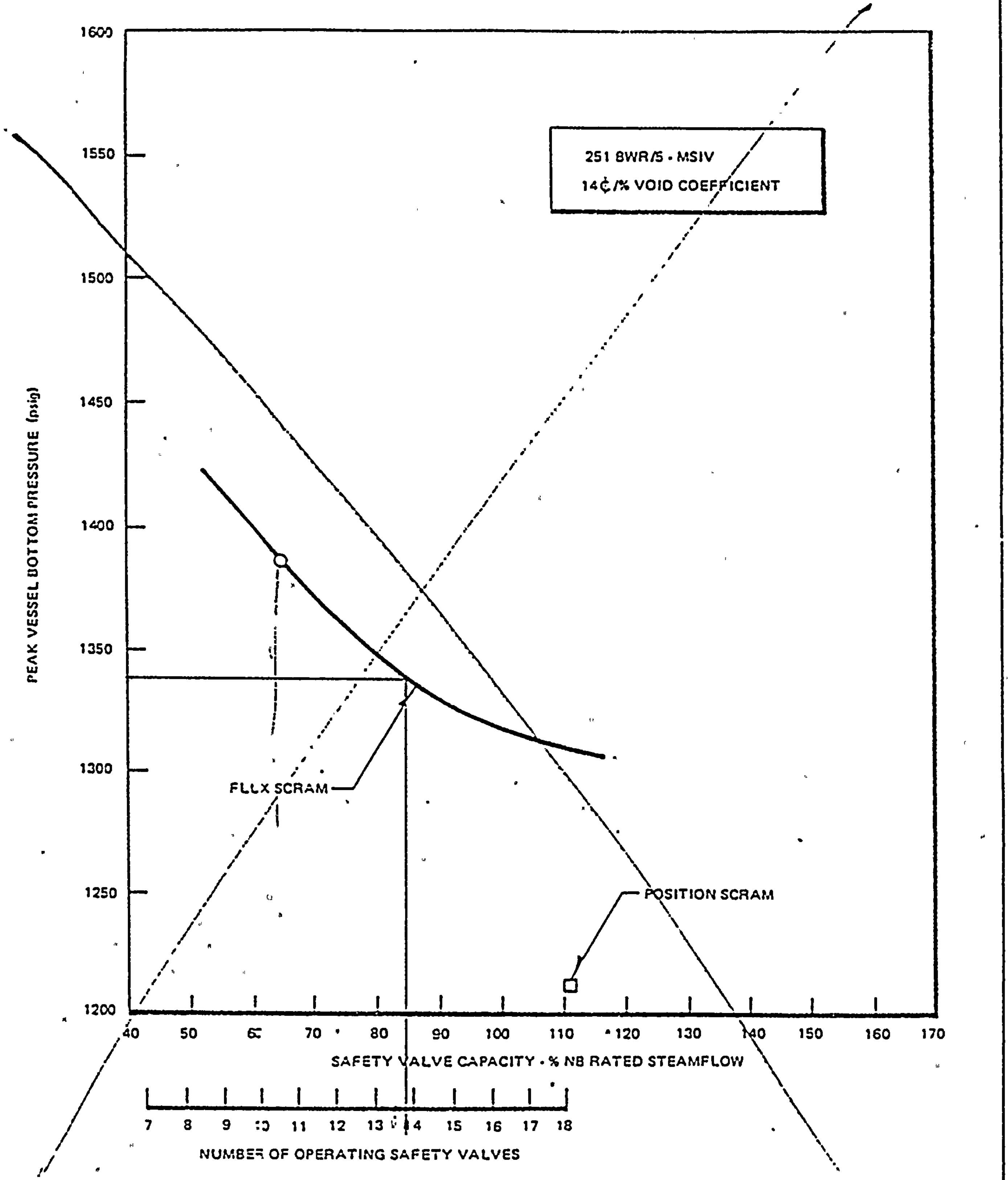
CIM 5481T



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL ROD DRIVE AND SCRAM REACTIVITY VERSUS TIME CHARACTERISTIC

FIGURE 5.2-3

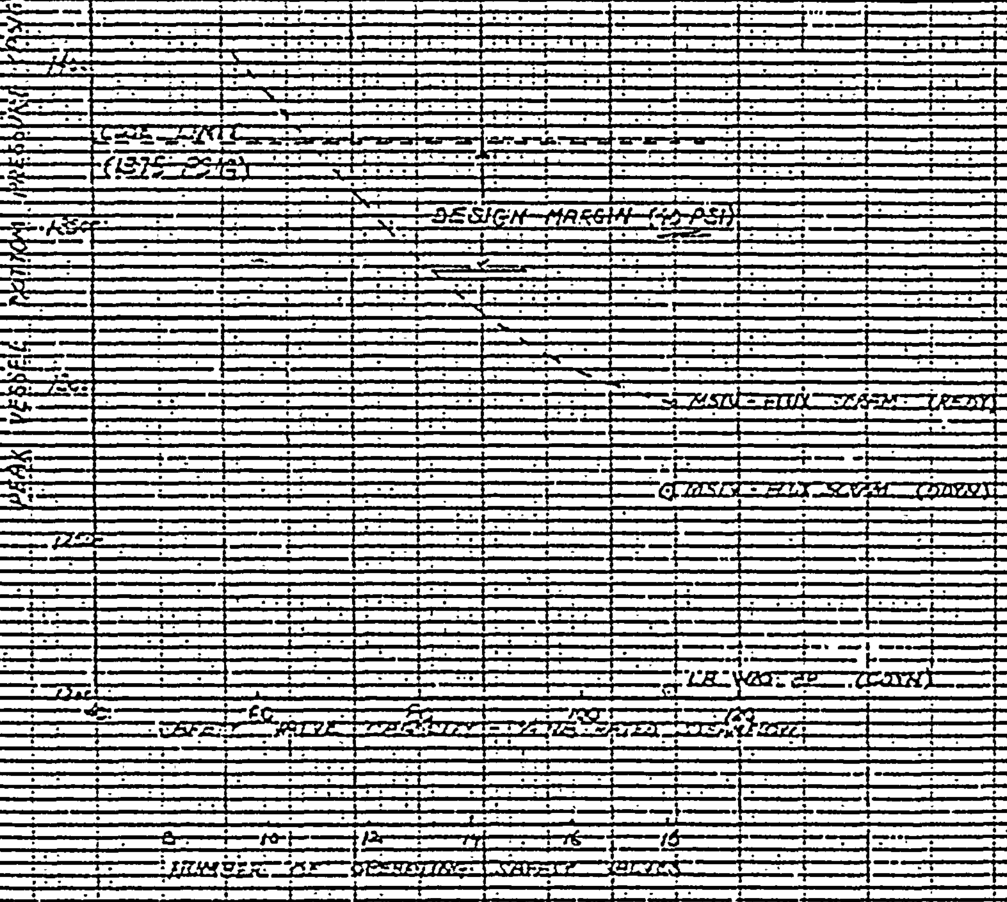


REPLACE WITH NEW FIGURE

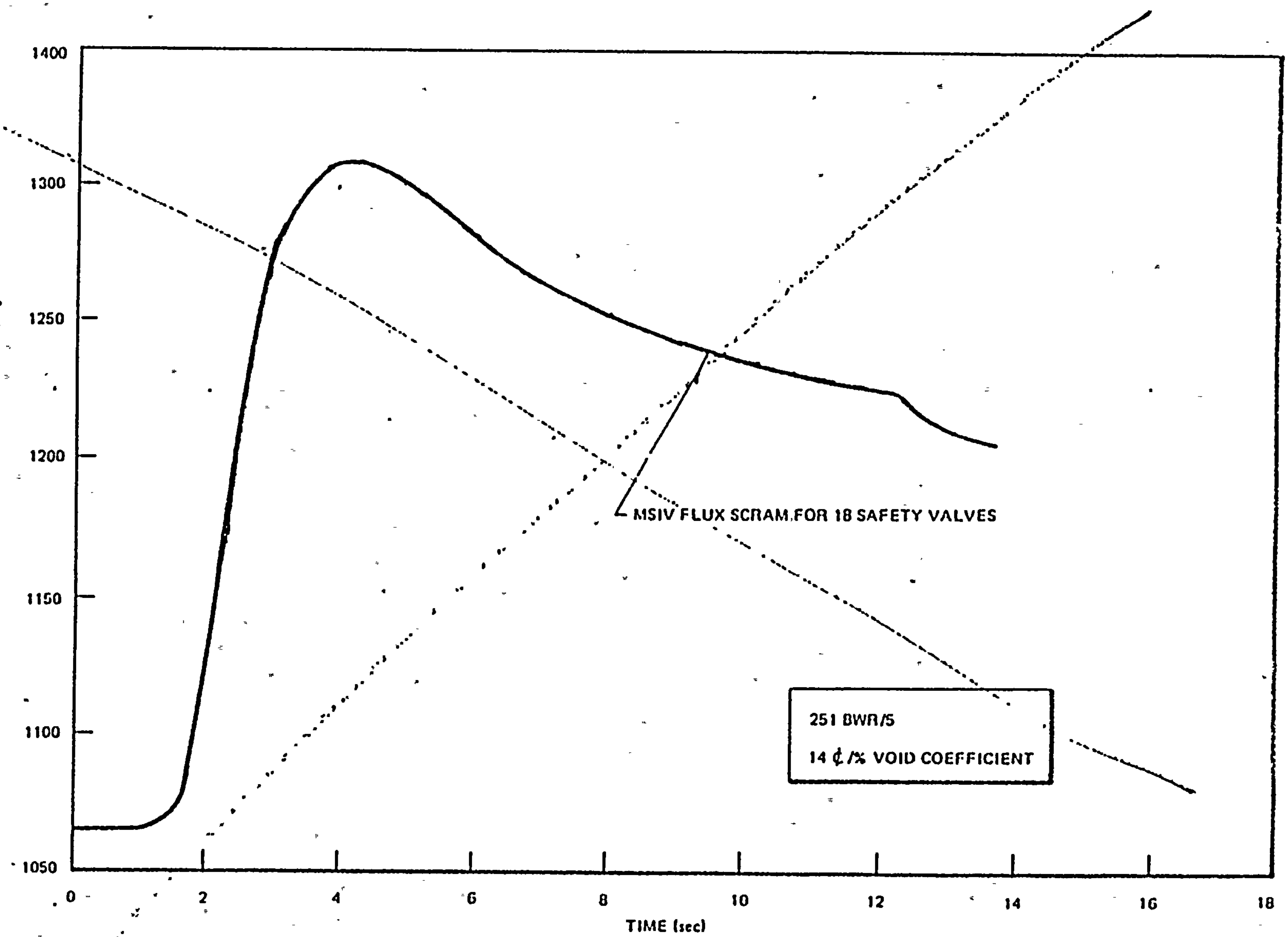
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PEAK VESSEL PRESSURE VERSUS SAFETY/RELIEF CAPACITY	FIGURE 5.2-4

FIGURE 5.2-4

PEAK VESSEL PRESSURE VERSUS SAFETY VALVE CAPACITY



REPLACE WITH NEW FIGURE



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

PEAK VESSEL PRESSURE VERSUS TIME FOR SAFETY VALVE CAPACITY SIZING TRANSIENT

FIGURE 5.2-5

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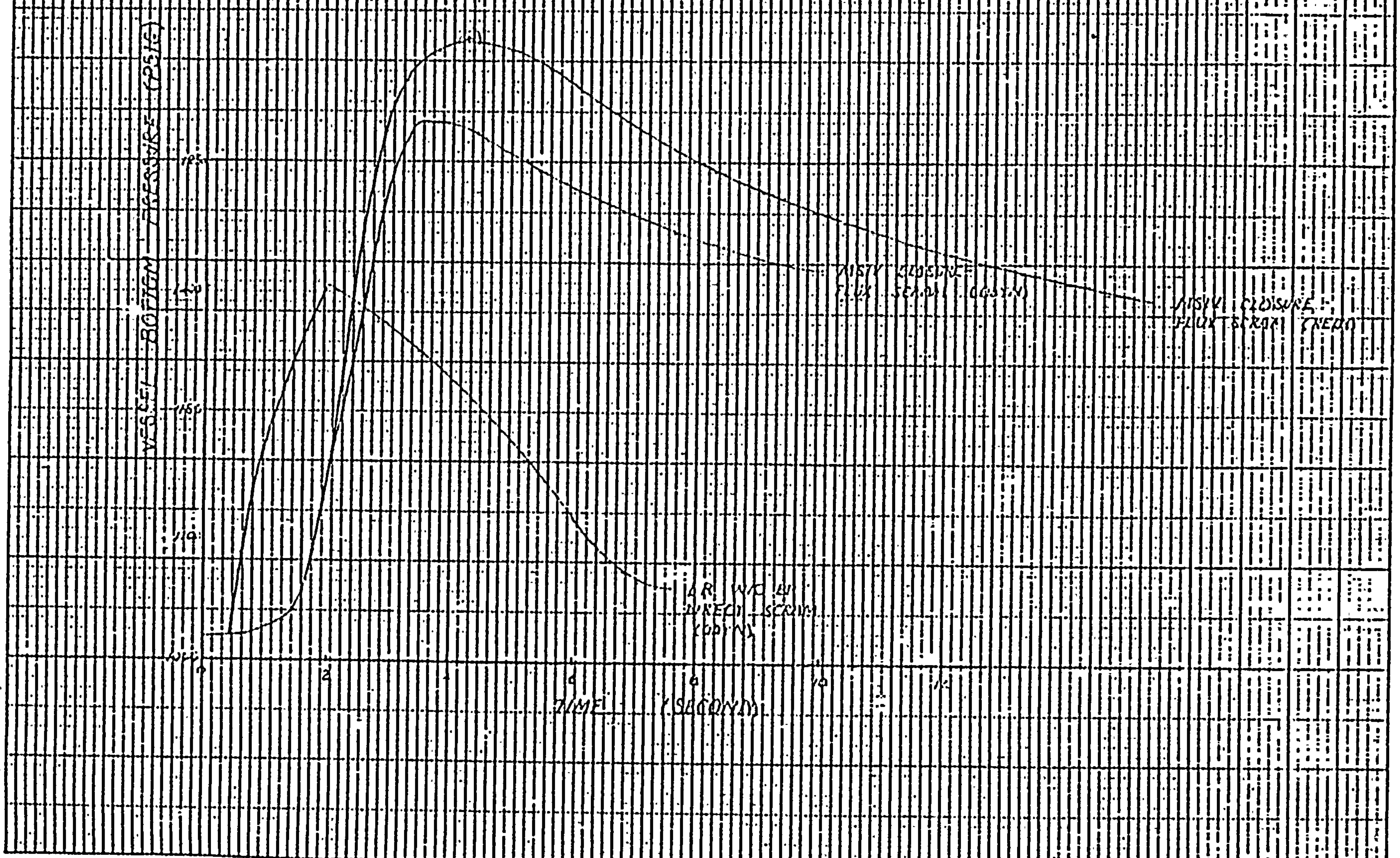
FIGURE 5.2-5
TIME RESPONSE OF PRESSURE VESSEL FOR PRESSURIZATION EVENTS



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TABLES

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FIGURES

15.0.3.2.1.3 Single Active Component Failure or Single Operator Error Analysis

a. The undesired action or maloperation of a single active component, or

b. Any single operator error where operator errors are defined as in 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

The various fuel failure mechanisms are described in 4.4, "Thermal and Hydraulic Design". Avoidance of unacceptable safety limits 1 and 2 (4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (see GETAB Reference 15.0-2). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than 1.06. ~~The reactor steady-state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter.~~ ←INSERT B

The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multi-node, single channel thermal-hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of Reference 15.0-2. ~~The initial condition assumed for all full power transient MCPR calculations is that the bundle is operating at both the linear heat generation rate limit (13.4 kW/ft) and at the MCPR limit (1.24).~~

INSERT B

Determination of the steady-state operating limit is accomplished as follows:

1) The change in the critical power ratio (ΔCPR) which would result in the safety limit CPR (1.06) being reached, is calculated for each event. These values are shown in Table 15.0-1.

2) The ΔCPR value is then added to the safety limit CPR value (1.06) to result in the event based MCPR except for events whose ΔCPR is calculated using ODYN.

3) For events whose ΔCPR is determined by ODYN (all rapid pressurization events) the event based MCPR is determined in conjunction with correction factors, the ΔCPR and the safety limit CPR. These correction factors are explained in detail in Section 3/4.2.3 of the Technical Specifications.

These results are given in Table 15.0-3 and Figure 15.0-3 for the limiting transients.

The operating limit MCPR is the maximum value of the

event MCPRs calculated from the transient analysis. The maximum calculated transient MCPR is depicted by the solid line in Figure 15.0-3. Maintaining the CPR operating limit at or above this operating limit assures that the safety limit CPR of 1.06 is never violated.

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2/4/81

TABLE 15.0-1

RESULTS SUMMARY OF TRANSIENT EVENTS APPLICABLE TO WNP-2

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	~~Minimum~~ ΔCPR*** -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blowdown	Duration of Blowdown: Duration of Blowdown sec
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	124.2	1030	1070	1001	117.1	0.16 ~~1.08~~	a		
15.1.2	15.1-3	Feedwater Cntl Failure, Max Demand	154.3 ~~176.0~~	1148 ~~1141~~	1177 ~~1170~~	1140 ~~1124~~	108.7 ~~110.6~~	0.08 (1) ~~1.09~~	a	14 ~~18~~	>20 ~~8.0~~
15.1.3	15.1-4	Pressure Regulator Fail-Open	104.3	1098	1117	1097	100.1	~~>1.18**~~ <0.06**		2	6.4
15.1.4		Inadvertent Opening of Safety or Relief Valve	SEE TEXT						a		
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temperature	SEE TEXT						a		
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Fail - Closed	SEE 15.2.2 and 15.2.3 W/Bypass on						a		
15.2.2	15.2-1	Generator Load Rejection, Bypass-On	156.8 ~~165.1~~	1145 ~~1137~~	1173 ~~1165~~	1140 ~~1122~~	102.9 ~~103.3~~	0.04 (1) ~~1.15~~	a	14 ~~18~~	~5.5 ~~5.5~~
15.2.2	15.2-2	Generator Load Rejection, Bypass-Off	236.4 ~~254.5~~	1173 ~~1165~~	1202 ~~1193~~	1168 ~~1148~~	107.8 ~~110.5~~	0.09 (1) ~~1.05~~	~~b~~ a	18	~7 ~~8.2~~
15.2.3	15.2-3	Turbine Trip, Bypass-On	147.5	1136	1163	1121	101.7	0.06 ~~1.18~~	a	18	5.5
15.2.3	15.2-4	Turbine Trip, Bypass-Off	218.3 ~~233.7~~	1173 ~~1163~~	1201 ~~1191~~	1168 ~~1147~~	106.4 ~~108.9~~	0.08 (1) ~~1.07~~	~~b~~ a	18	~7 ~~8.2~~
15.2.4	15.2-5	Inadvertent MSIV Closure	186.2	1154	1191	1146	100.0	<0.04** ~~>1.20**~~	a	18	5.7
15.2.5	15.2-6	Loss of Condenser Vacuum	157.5	1135	1162	1120	102.6	<0.17** ~~≥1.07**~~	a	18	5.4
15.2.6	15.2-7	Loss of Auxiliary Power Transformers	104.3	1153	1172	1149	100.0	~0.00** ~~>1.24**~~	a	0	0

TABLE 15.0-1 - (Con. ued)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	~~Minimum~~ ΔCPR*** -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blowdown	Duration of Blowdown: Duration of Blowdown sec
15.2.6	15.2-8	Loss of All Grid Connections	144.2	1157	1177	1137	101.6	<0.09** ~~>1.15**~~	a	10	5.5
15.2.7	15.2	Loss of all Feedwater Flow	104.3	1095	1107	1095	100.0	~0.00** ~~>1.24**~~	a	2	6.0
15.2.8		Feedwater Piping Break	SEE 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	SEE TEXT								
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104.4	1021	1061	994	100.0	~0.00** ~~>1.24**~~	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104.4	1104	1116	1100	100.1	~0.00** ~~>1.24**~~	a	6	5.3
15.3.2	15.3-3	Fast Closure of One Main Recirc Valve	104.3	1101	1115	1097	100.0	~0.00** ~~>1.24**~~	a	2	6.8
15.3.2	15.3-4	Fast Closure of Two Main Recirc Valves	104.4	1105	1115	1100	100.0	~0.00** ~~>1.24**~~	a	6	5.4
15.3.3	15.3-5	Seizure of One Recirculation Pump	104.3	1105	1117	100	100.2	~0.00** ~~>1.24**~~	c	6	5.4
15.3.4		Recirc Pump Shaft Break	SEE 15.3.3								
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling	SEE TEXT						b		
15.~~5~~4.1.2		RWE - Startup	SEE TEXT						b		
15.4.2		RWE - At Power	SEE TEXT					~~1.17~~	a		
15.4.3		Control Rod Misoperation	SEE 15.4.1 and 15.4.2								
15.4.4	15.4-6	Abnormal Startup of Idle Recirculation Loop	94.2	981	995	970	146.6	~0.00** (a) ~~>1.06**~~	a		

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TABLE 15.0-1 - (Continued)

Paragraph I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	~~Minimum~~ ΔCPR*** -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blowdown	Duration of Blowdown: Duration of Blowdown sec
15.4.5	15.4-7	Fast Opening of One Main Recirc Valve	282.9	980	1000	971	141.0	~0.00** (2) ≥1.06**	a		
15.4.5	15.4-8	Fast Opening of Both Main Recirc Valves	222.2	977	1000	969	134.6	~0.00** (2) ~~≥1.06**~~	a		
15.4.7		Misplaced Bundle Accident	SEE TEXT					~~1.12~~	b		
15.4.9		Rod Drop Accident							c		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104.3	1020	1061	993	100.1	~0.00** ~~~1.24**~~	a		
15.5.3		BWR Transients	SEE APPROPRIATE EVENTS IN 15.1 and 15.2								

* a: Moderate Frequency
b: Infrequent Incident
c: Limiting Fault

** Estimated Value

*** For Conservatism, ΔCPR is Based on an initial CPR which yields an MCPR = 1.06

(1) ODYN Results without the adjustment Factors delineated in the ODYN Report NEDO-24154, NEDE-24154P

(2) Event starts at an initial CPR which is greater than the OLMCPR. Resulting MCPR for the event approaches OLMCPR [illegible]

TABLE 15.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

	REDY	ODYN
1. Thermal Power Level, MWt		
Warranted Value	3323	3323
Analysis Value	~~3468~~ 3464	3468 (1)
2. Steam Flow, lbs per hr		
Analysis Value	14.98×10^6	14.98×10^6
3. Core Flow, lbs per hr	108.36×10^6	108.5×10^6 (1)
4. Feedwater Flow Rate, lb per sec		
Analysis Value	4161	4161
5. Feedwater Temperature, ^oF	424	424.
6. Vessel Dome Pressure, psig	1020	1020
7. Vessel Core Pressure, psig	1031	1031
8. Turbine Bypass Capacity, % NBR	25	25
9. Core Coolant Inlet Enthalpy,		
Btu per lb	529.3	529.3
10. Turbine Inlet Pressure, psig	975	975
11. Fuel Lattice	8 x 8	P8X8R (2)
12. Core Average Fuel Cladding Gap		
Conductance, Btu/sec-ft^2-^oF	0.1667	0.1744 (3)
13. Core Leakage Flow, %	11.84	11.84
14. Required MCPR Operating Limit	1.24 (4)	1.24 (4)
15. MCPR Safety Limit	1.06	1.06
16. Doppler Coefficient (-)¢/^oF		
Nominal EOC-1	0.227	0.227
Analysis Data	0.215	0.215

TABLE 15.0-2 - (Continued)

		REDY	CDYN
17.	Void Coefficient (-)¢/% Rated Voids		
	Nominal EOC-1	7.48	7.48
	Analysis Data for Power Increase Events	12.70	(5)
	Analysis Data for Power Decrease Events	7.065	(5)
18.	Core Average Rated Void Fraction, % (STEADY STATE)	41.32	41.37 (1)
19.	Scram Reactivity, $\$\Delta k$ Analysis Data	Figure 15.0-2	(5)
20.	Control Rod Drive Speed, Position versus time	Figure 15.0-2	Figure 15.0-2
21.	Jet Pump Ratio, M	2.41	2.41
22.	Safety/Relief Valve Capacity, % NBR		
	SAFETY VALVE CAPACITY @ 1213 psig	111.5	111.5
	RELIEF VALVE CAPACITY @ 1106 psig	101.8	101.8
	Manufacturer	Crosby	CROSBY
	Quantity Installed	18	18
23.	Relief Function Delay, seconds	0.4	0.4
24.	Relief Function Response, seconds	0.1	0.1
25.	Set Points for Safety/Relief Valves		
	Safety Function, psig	1177, 1187, 1197, 1207, 1217	1177, 1187, 1197, 1207, 1217
	Relief Function, psig	1091, 1101, 1111, 1121, 1131	1106, 1116, 1126 (1) 1136, 1146
26.	Number of Valve Groupings Simulated		
	Safety Function, No.	5	5
	Relief Function, No.	5	5
27.	High Flux Trip, % NBR		
	Analysis set point (12~~0~~1 x 1.043), % NBR	126.20 ~~125.16~~	126.20

TABLE 15.0-2 (Continued)

		REDY	ODYN
28.	High Pressure Scram Set Point, psig	1071	1071
29.	Vessel Level Trips, Feet Above DRYER ~~Separator~~ Skirt Bottom		
	Level 8 - (L8), ~~feet~~ inches	~~5.750~~ 55.5	59.5 (1)
	Level 4 - (L4), ~~feet~~ inches	~~3.750~~ 31.5	(6)
	Level 3 - (L3), ~~feet~~ inches	~~2.167~~ 12.5	(6)
	Level 2 - (L2), ~~feet~~ inches	~~(-)2.041~~ (-) 33	(6)
30.	APRM Thermal Trip		
	Set Point, % NBR @ 100% core flow	122.030	122.030
31.	Recirculation Pump Trip Delay, Seconds	0.140	0.190 (1)
32.	Recirculation Pump Trip Inertia time constant for analysis, seconds	6 (7)	6 (7)

NOTES:

(1) Corrected to reflect the latest design data.

(2) P8X8R: Prepresurized, retrofit fuel Pins in 8X8 array.

(3) Increase in conductance (from Redy) is due to the initial prepresurization of the fuel Pins.

(4) See Table 15.0-3

(5) ODYN VALUES ARE calculated within the code for end of Cycle 1 condition.

(6) Parameter not used in the ODYN ANALYSIS.

(7) The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_o n}{g T_o}$$, where

t = inertia time constant (Sec).

J_o = pump motor inertia ($lb\text{-}ft^2$).

n = rated pump speed (rps).

g = gravitational constant (ft/sec^2).

T_o = pump shaft torque (lb-ft).

GENERAL ELECTRIC

NUCLEAR ENERGY DIVISIONS

TABLE 15.0-3

REQUIRED OPERATING LIMIT CPR VALUES

Pressurization Events:

	CPR (Option A)*	CPR (Op
Load Rejection Without Bypass	1.20	1.12
Turbine Trip Without Bypass	1.19	1.11
Feedwater Controller Failure	1.19	1.16
Load Rejection	1.15	1.07

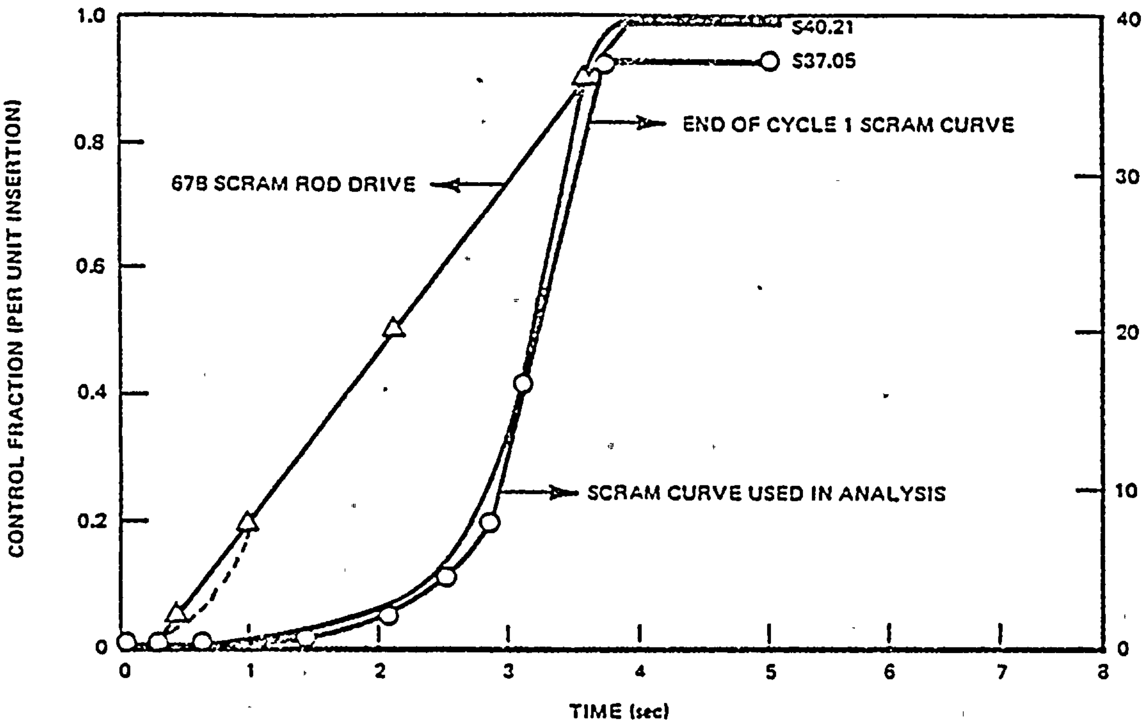
Non-Pressurization Events:

	CPR
Rod Withdrawal Error	1.24**
Loss of Feedwater Heater	1.22

---

* Includes adjustment factors as specified in the NRC safety evaluation report on ODYN, NEDO-24154 and NEDE-24154P

** Required OLCPR using either Option A or Option B adjustment factor regardless of frequency category of the turbine-generator trip events with bypass failure.



--- Revised portion of scram speed used in ODYN analysis

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SCRAM POSITION AND REACTIVITY CHARACTERISTICS	FIGURE 15.0-2

HANFORD 2

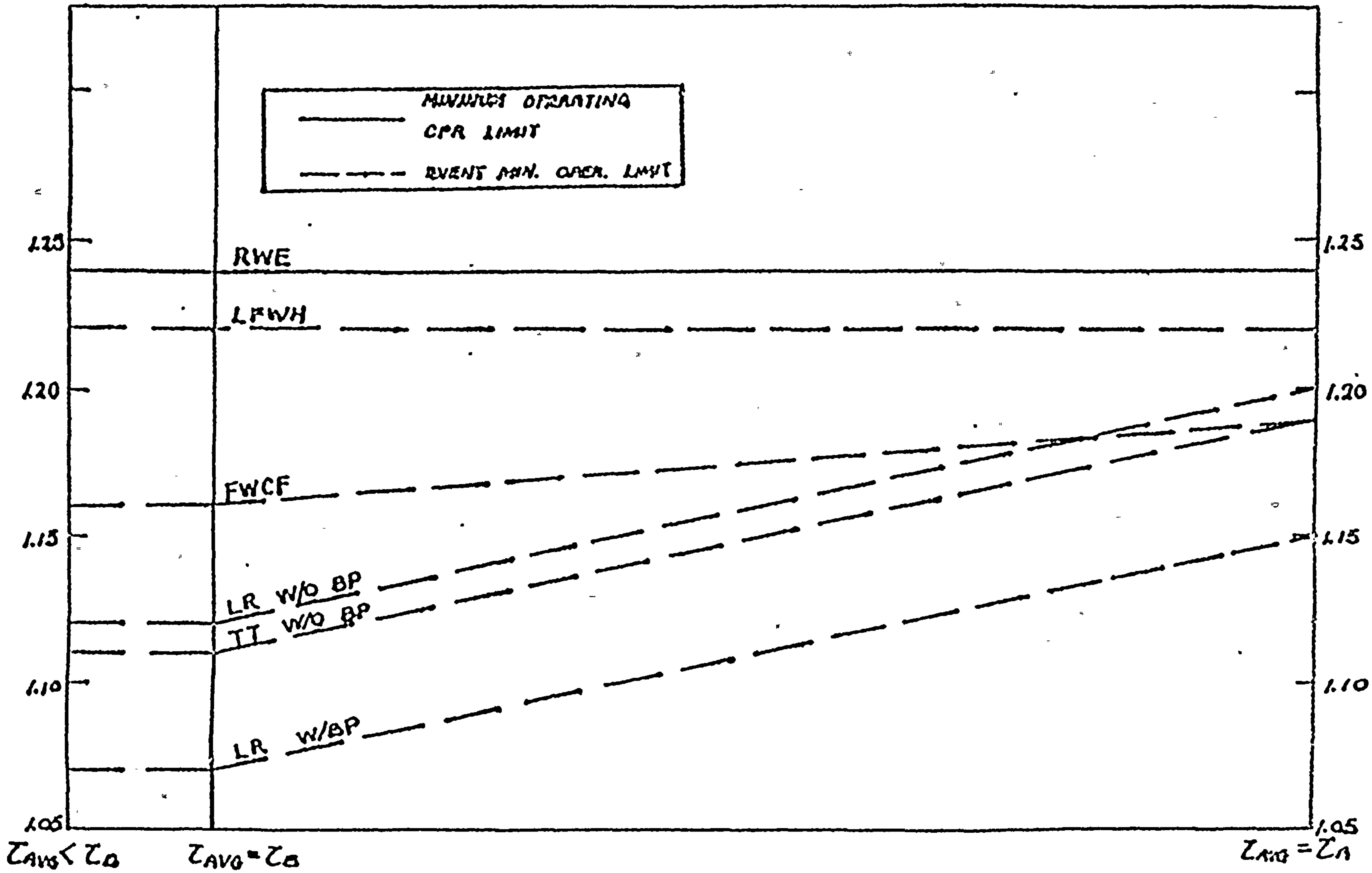


FIG. 15.0-3 Minimum Operating CPR Limit vs. Scram Speed

reactor core isolation cooling system and the high pressure core spray system to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Table 15.1-3 the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 set point. At this point in the logic a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission however is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature." However, high moisture entering the turbine will be detected by high levels in the turbine's moisture separators which trip the unit. In addition, excessive moisture entering the turbine will cause vibration to the point where it too will trip the unit.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. See 15A for a detailed discussion of this subject.

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

~~The computer model described in 15.1.2.3.1 was used to simulate this event.~~ ← INSERT C

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-2.

~~The same void reactivity coefficient used for pressurization transients is applied since a more negative value conservatively increases the apparent severity of the power increase.~~ End of cycle (all rods out) scram characteristics are assumed. The safety-relief valve action is conservatively assumed to occur with higher than nominal set points. The transient is simulated by programming an upper limit failure in the feedwater system such that 135% feedwater flow occurs at the design pressure of 1060 psig.

INSERT C

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 15.1-3. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear computer simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

a. An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks.

b. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

c. The steam lines are modeled by eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transient.

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure and load demand, are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

g. The control systems and reactor protection system models are, for the most part, identical to those employed in the point reactor model, which is described in detail in Reference 15.1-1 and used in analysis for other transients.

15.1.2.3.3 Results

The simulated feedwater controller transient at 105% NBR steam flow is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 14.4 ~~10.2~~ sec. Scram occurs simutaneously from stop valve closure, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

~~The MCPR reaches 1.09 and peak fuel center temperature increase 322°F.~~ The peak neutron flux is 154.3% NBR and the maximum surface heat flux is 108.7% of initial condition. The turbine bypass system and the main steam safety/relief valves open to limit the peak vessel bottom pressure to 1177 ~~1170~~ psig, and the nuclear system process barrier pressure limit is not endangered. The bypass valves subsequently close to re-establish pressure control in the vessel during shutdown. ~~The level will gradually drop to the low level isolation reference point, activating the RCIC/HPCS systems for long term level control.~~ INSERT D

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative response (e.g., relief setpoints, scram stroke time and work characteristics). Expected plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under optimum meteorological and release conditions. If purging of the containment is chosen the release will be in accordance with established technical specifications; therefore, this event, at worst, would only result in a small increase in the yearly intergrated exposure level.

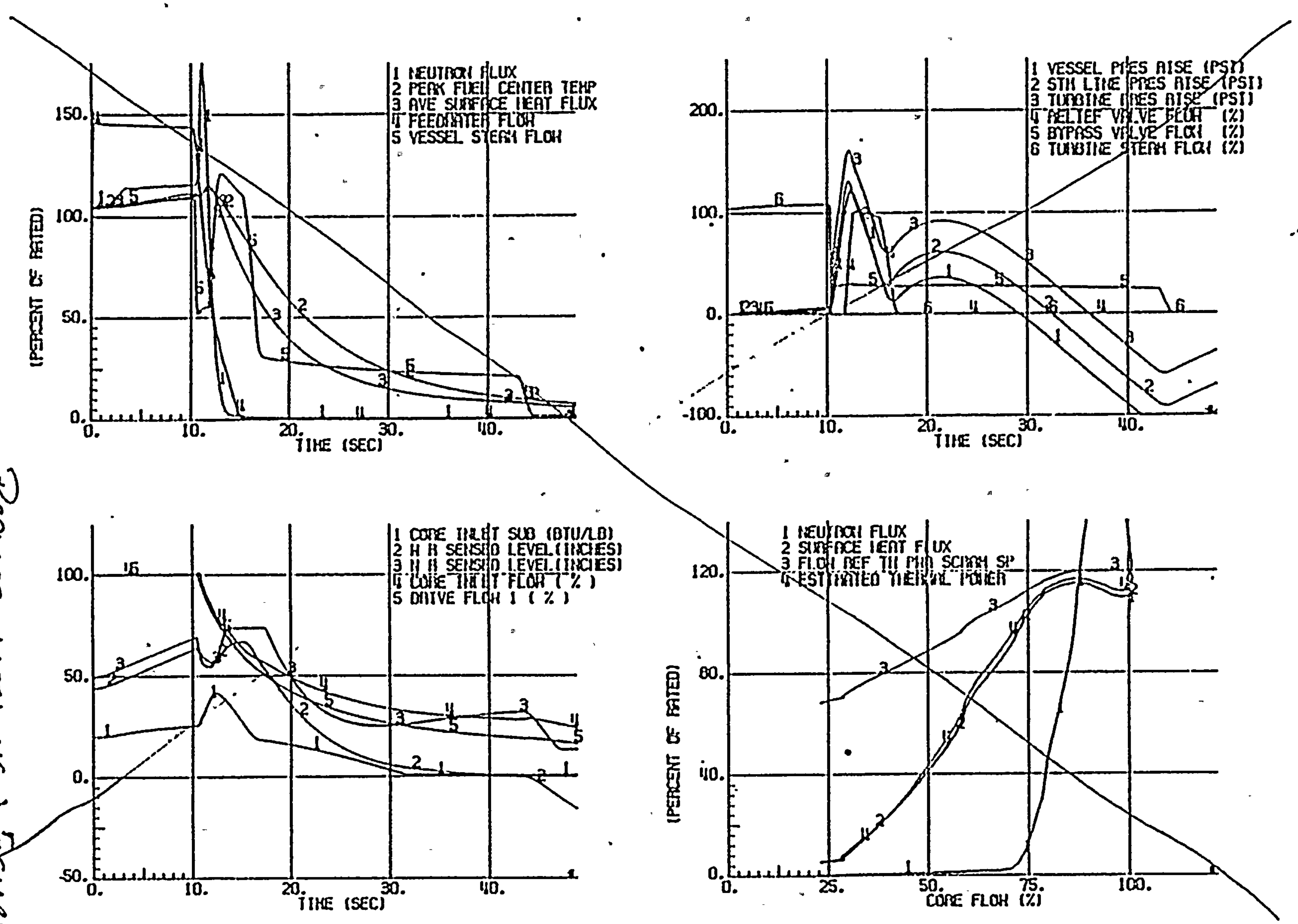
INSERT D

Events caused by low water level trips, including ~~tripping of recirculation system pumps,~~ closure of main steam line isolation valves, and initiation of HPCS ~~HPCI~~ and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

TABLE 15.1-3

SEQUENCE OF EVENTS FOR FIGURE 15.1-3

Time-sec	Event
0	Initiate simulated failure of 146% upper limit on feedwater flow.
11.65	Turbine bypass valves start to open.
~~10.24~~ 14.34	L8 vessel level set point trips main turbine and feedwater pumps. Turbine bypass operation initiated.
~~10.25~~ 14.35	Reactor scram trip actuated from main turbine stop valve position switches.
~~10.25~~ 14.35	Recirculation pump trip (RPT) actuated by stop valve position switches.
~~10.24~~ 14.45	Main turbine stop valves closed ~~and turbine bypass valves start to open~~.
~~10.38~~ 14.54	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
15.90 - 16.63 ~~11.65-12.33~~	Group 1-~~5~~ 4 relief valves actuated due to high pressure.
~~15.4 (est.)~~	~~Group 5 relief valves start to close.~~
~~17.5 (est.)~~	~~All relief groups closed.~~
~~43.1 (est.)~~	~~Turbine bypass valves start to close.~~
~~44.4~~	~~Turbine bypass valves closed.~~
~~50+~~	~~Main steam line isolation and RCIC and HPCS systems initiation on wide range low level (L2). Relief Group 1 cycles open and close on pressure.~~



15.1-15

REPLACE WITH NEW FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

FEEDWATER CONTROLLER FAILURE, MAX DEMAND

FIGURE 15.1-3

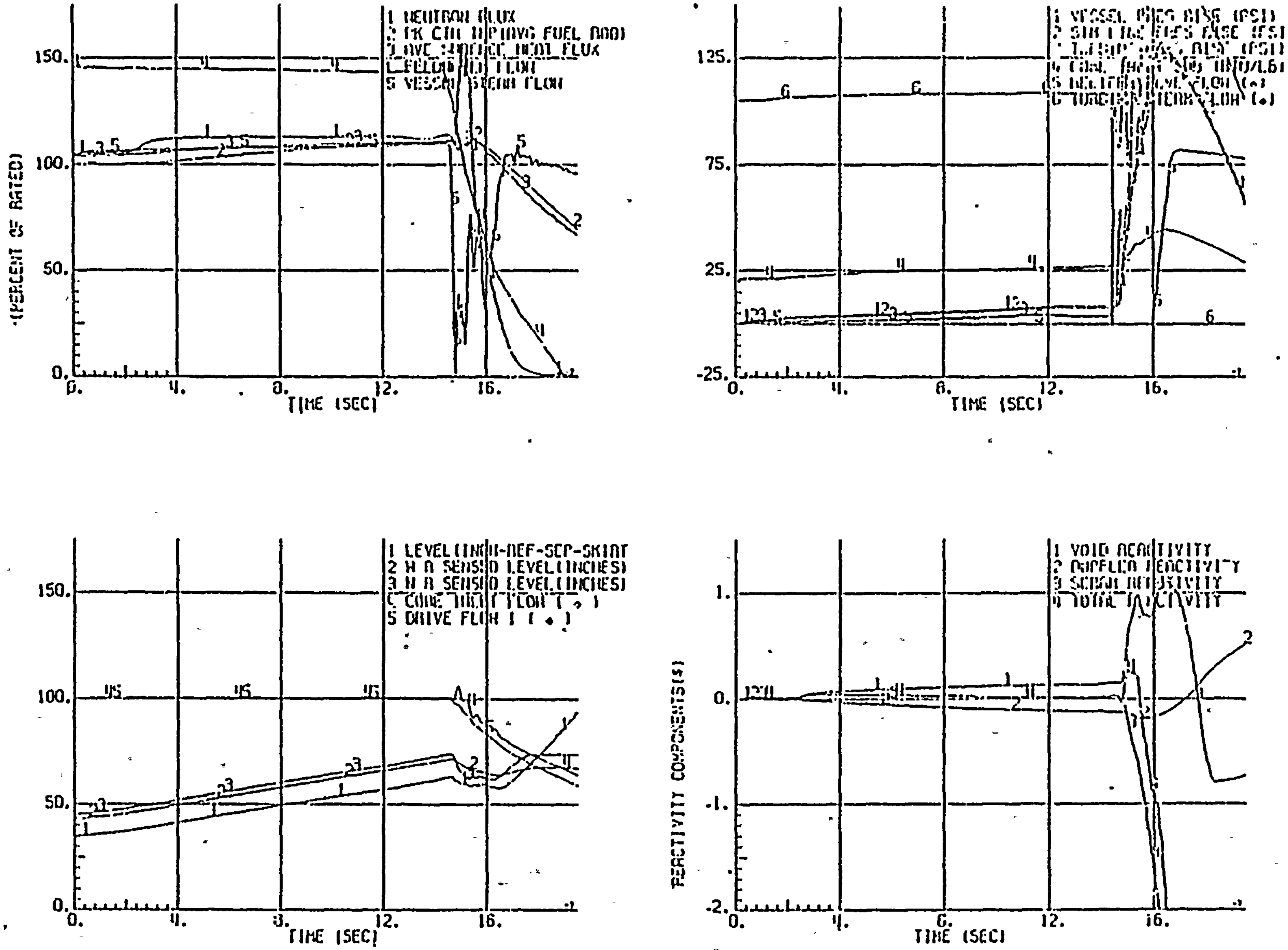


FIG. 15.1-3
KKICHN01FX103
FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND, WITH HIGH WATER LEVEL TRIPS

15.1.7 REFERENCES

15.1-1 R.B. Linford, "Analytical Methods ... Evaluations for the General Electri... Reactor," April 1973 (NEDO 10802).

15.1-2 "Plant Design Assessment Report for LOCA ... (Revision 2)," Washington Public Power Sup... transmitted to NRC as Amendment No. 6 to the ... September 19, 1979.

15.1-3 F. ODAR, "Safety Evaluation for General Electric Topical Report: Qualification of the one-dimensional core transient model for Boiling Water Reactors," NEDO-24154, NEDE-24154P, Volumes 1, 2 and 3, 1980.

15.2.2 GENERATOR LOAD REJECTION

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (T-G) rotor. Closure of the main turbine control valves will cause a sudden reduction in steam flow which results in an increase in system pressure and reactor shutdown.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection

This event is categorized as an incident of moderate frequency.

15.2.2.1.2.2 Generator Load Rejection with Bypass Failure

This event is categorized as ~~an infrequent incident~~ a moderate frequency event with the following characteristics:

Frequency: 0.0036/plant year

MTBE: 278 years

Frequency Basis: Thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection - Turbine Control Valve Fast Closure

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-1.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Turbine control valve trip scram and recirculation pump trip (RPT) are designed to satisfy the single failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single failure analysis can be found in 15A.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The computer model described in ~~15.1.1.3.1~~ 15.1.2.3.1 was used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-2.

The turbine digital electrohydraulic control system (DEH) power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of .15 seconds.

Auxiliary power would normally be independent of any turbine-generator overspeed effects and continuously supplied at rated frequency as automatic fast transfer to auxiliary power supplies normally occurs. For the purposes of worse case analysis, the recirculation pumps are assumed to remain tied to the main generator and thus increase in speed with the T-G overspeed until tripped by the recirculation pump trip system (RPT).

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

INSERT E

~~Although the closure of main steam isolation valves as caused by low water level trip (L2) is included in the simulation, the flows from initiation of RCIC and HPCS core cooling system functions are not included as normal startup and actuation can take up to 30 seconds before effects are realized. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system.~~

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15.2-1 shows the results of the generator trip from rated power. Peak neutron flux rises ~~165%~~ 156.8% above NB rated conditions.

The average surface heat flux peaks at ~~102.3%~~ 102.9% of the initial value and MCPR does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

Figure 15.2-2 shows that, for the case of Bypass failure, peak neutron flux reaches about ~~254.4%~~ 236.4% of rated, average surface heat flux reaches ~~110.5%~~ 107.8% of its initial value. ~~Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria and the MCPR limit is permitted to fall below the safety limit for the incidents of moderate frequency. MCPR reaches 1.05 for this event.~~

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure time of the turbine control valve of .15 seconds is conservative. Typically, the actual closure time is more like .2 seconds. Clearly the less time it takes to close, the more severe the pressurization effect.

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief

INSERT E

Events caused by low water level trips, including ~~tripping of recirculation system pumps,~~ closure of main steam line isolation valves, and initiation of HPCS ~~HPCI~~ and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

set points, scram stroke time and work characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal operating range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the valves reaches ~~1148~~ 1168 psig. The peak nuclear system pressure reaches ~~1183~~ 1202 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While the consequences of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under optimum meteorological and release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specifications; therefore, this event, at worst, would only result in a small increase in the yearly integrated exposure level.

TABLE 15.2-1

SEQUENCE OF EVENTS FOR FIGURE 15.2-1

Time-sec	Event
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0	Fast control valve closure initiates scram trip.
0	Fast control valve closure initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
~~0.10~~ 0.11	Turbine bypass valves start to open.
~~0.14~~ 0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
~~1.56~~ 1.70	Group 1 relief valves actuated.
~~1.71~~ 1.86	Group 2 relief valves actuated.
~~1.87~~ 2.01	Group 3 relief valves actuated.
~~2.06~~ 2.27	Group 4 relief valves actuated.
~~2.36~~	~~Group 5 relief valves actuated.~~
~~4.63~~	~~Feedwater turbines trip on L8 high water level.~~

TABLE 15.2-1 - (Continued) Page 2 of 2

Time-sec	Event
~~5.2 (est.)~~	~~Group 5 pressure relief valves start to close.~~
~~7.2 (est.)~~	~~All relief groups closed.~~
~~31.5~~	~~Turbine bypass starts to close.~~
~~32.4 (est.)~~	~~Turbine bypass closed.~~
~~40.0~~	~~Turbine bypass reopens on pressure increase at turbine.~~
~~42.2~~	~~Main steam line isolation and RCIC and HPCS systems initiation on low level (L2).~~
~~50+~~	~~Group 1 pressure relief valves cycle open and close on pressure.~~

TABLE 15.2-2 Page 1 of 2

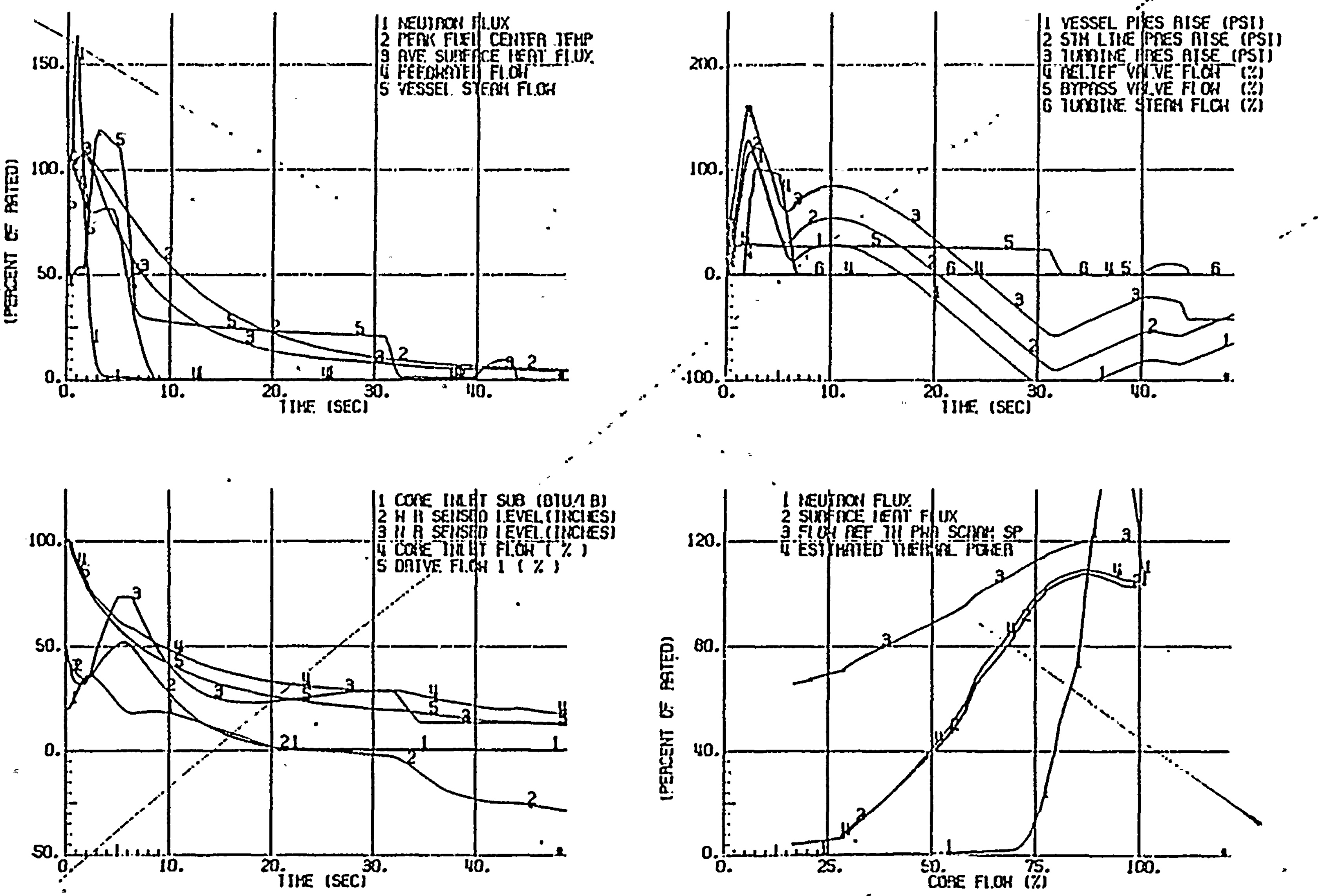
SEQUENCE OF EVENTS FOR FIGURE 15.2-2

Time-sec	Event
(-)0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure initiates scram trip.
0	Fast control valve closure initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
~~0.14~~ 0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
~~1.30~~ 1.35	Group 1 relief valves actuated.
~~1.40~~ 1.48	Group 2 relief valves actuated.
~~1.51~~ 1.61	Group 3 relief valves actuated.
~~1.61~~ 1.73	Group 4 relief valves actuated.
~~1.73~~ 1.86	Group 5 relief valves actuated.
~~7.08~~	~~Feedwater turbines trip on L8 high water level.~~
~~7.20 (est.)~~	~~Group 5 pressure relief valves start to close.~~

TABLE 15.2-2 - (Continued) Page 2 of 2

Time-sec	Event
~~9.6 (est.)~~	~~All relief groups closed.~~
~~11.74~~	~~Group 1 relief valves reactuated on high pressure.~~
~~24.7 (est.)~~	~~Relief group 1 closed.~~
~~35.64~~	~~Group 1 relief valves reactuated on high pressure.~~
~~35.83~~	~~Main steam line isolation and RCIC and HPCS systems initiation on low level (L2).~~
~~41.7 (est.)~~	~~Relief group 1 closed.~~
~~50+~~	~~Group 1 pressure relief valves cycle open and close on pressure.~~

REPLACE WITH NEW FIGURE 15.2-13



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

GENERATOR LOAD REJECTION, TRIP SCRAM,
BYPASS AND RPT - ON

FIGURE
15.2-1

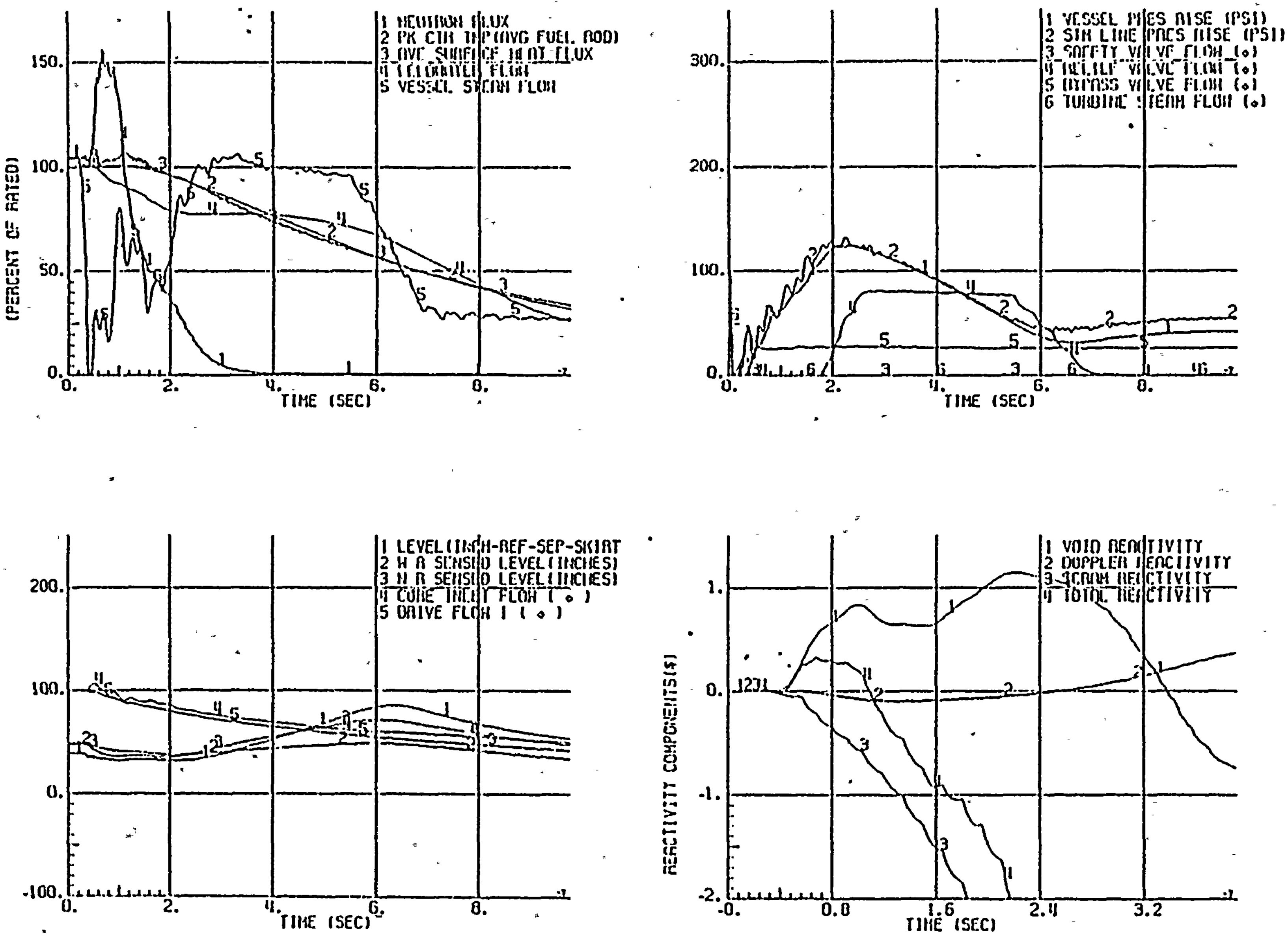
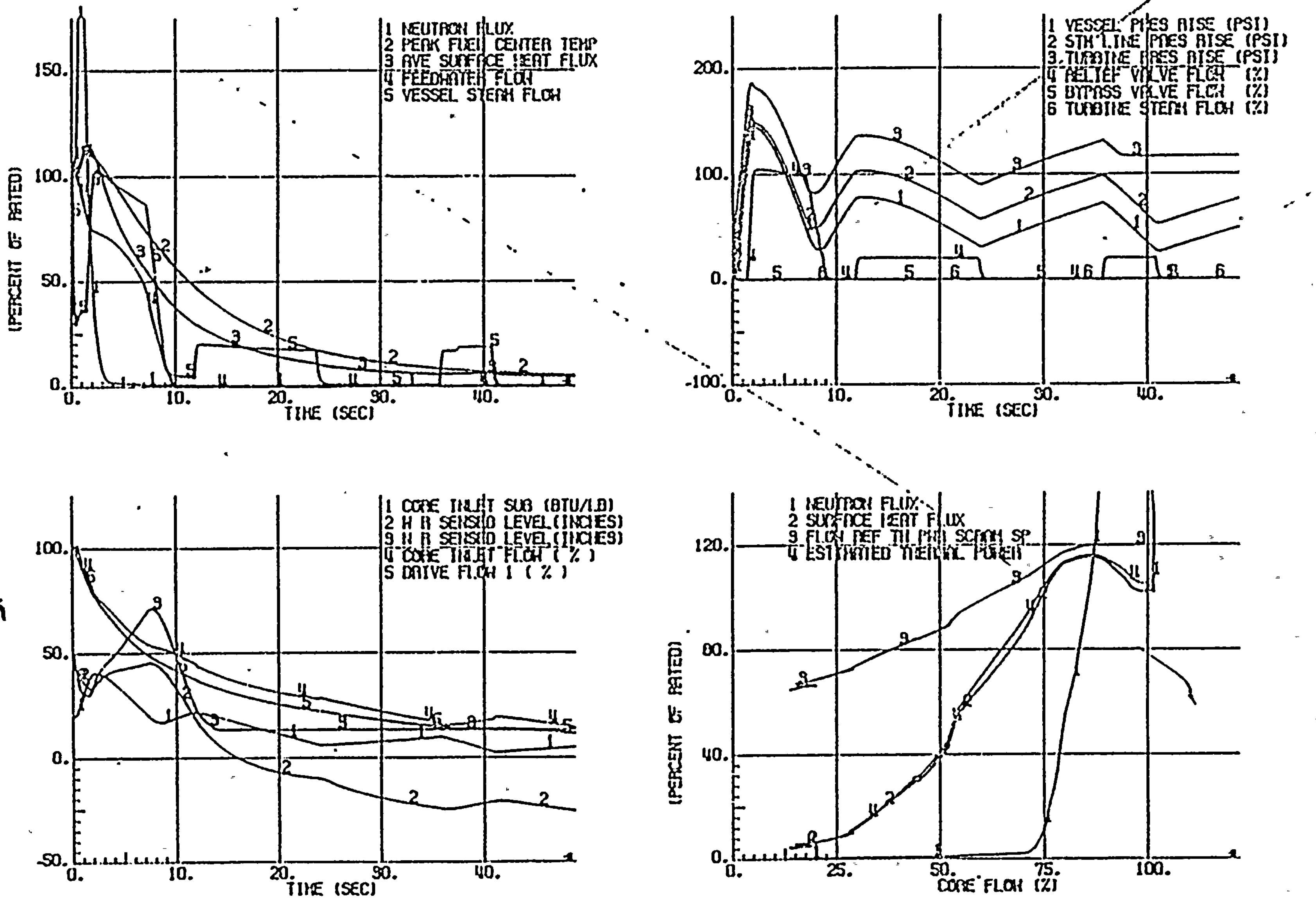


FIG. 15.2-1
KK1EHM01LR10G..
GENERATOR LOAD REJECTION, WITH BYPASS ON, RPT-ON.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REPLACE WITH NEW FIGURE 15.2-14



GENERATOR LOAD REJECTION, TRIP SCRAM, BYPASS - OFF, RPT - ON

FIGURE 15.2-2

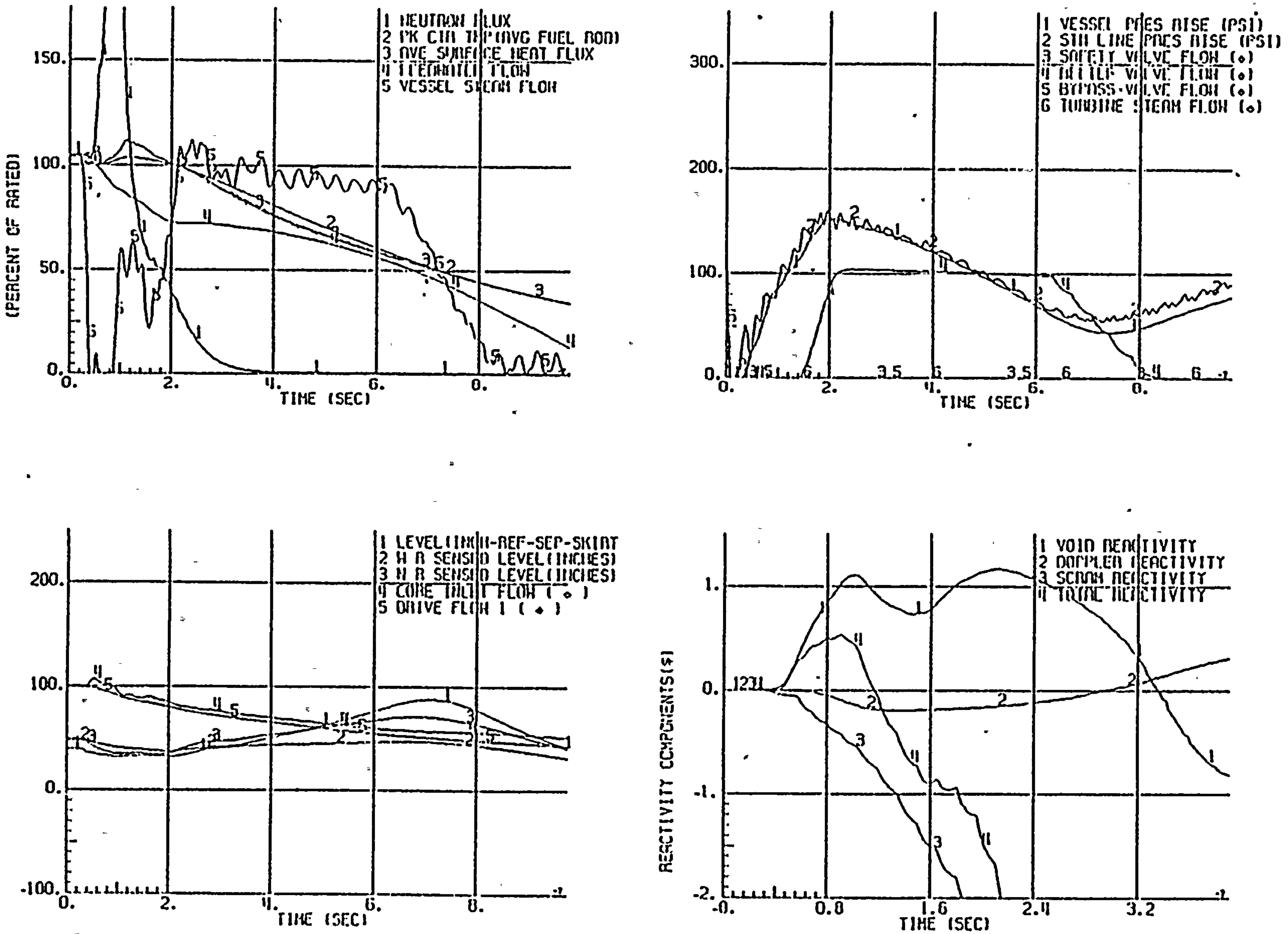


FIG. 15.2-2
KK1CHMOILR10S
GENERATOR LOAD REJECTION, WITHOUT BYPASS, RPT-ON

33271
181...2.

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lock out, loss of control fluid pressure, low condenser vacuum and reactor high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a by-product of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. In order to get an accurate event-by-event frequency breakdown, this type of division of initiating causes is required.

15.2.3.1.2.2 Turbine Trip with Failure of the Bypass

This transient disturbance is categorized as a~~n~~ ~~infrequent~~ moderate
Frequency ~~incident~~. Frequency is expected to be as follows:

Frequency: .0064/plant year

MTBE: 156 years

Frequency Basis: As discussed in 15.2.2.1.2.2, Generator Load Rejection with Bypass Failure, the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.33 events/plant year yields the frequency of .0064/plant year.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip

Turbine trip at high power produces the sequence of events listed in Table 15.2-3.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 30% NBR

Same as 15.2.3.2.3.1 except RPT and stop valve closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in 15.1.1.3.1 was used to simulate ~~these events.~~ the turbine trip with bypass event, and the model in Subsection 15.1.2.3.1 was used for the turbine trip with failure of bypass event.

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

Turbine stop valves full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the stop valves when the valves are 90% open or less. This stop valve scram trip signal is automatically bypassed when the reactor is below 30% NB rated power level.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105% NB rated steam flow conditions in Figure 15.2-3.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 147.5% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed 101.7% of its initial value. ~~MCPR for the transient is 1.18.~~

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105% NB rated steam flow conditions in Figure 15.2-4.

Peak neutron flux reaches ~~[illegible]~~ 218.3% of its rated value, and peak ~~fuel center temperature increases approximately 203°F. Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria and the MCPR limit is permitted to fall below the safety limit for incidents of moderate frequency. However, the MCPR for this transient is 1.07 which is just above the safety limit for incidents of moderate frequency and, therefore, the design basis is satisfied.~~

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 30% of rated power, the turbine stop valve closure and turbine control valve closure scrams are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

Surface heat flux reaches 106.4% of its initial value.

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves include errors and uncertainties (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1163 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1136 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches ~~1191~~ 1201 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed ~~1163~~ 1173 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since

TABLE 15.2-4 Page 1 of 2

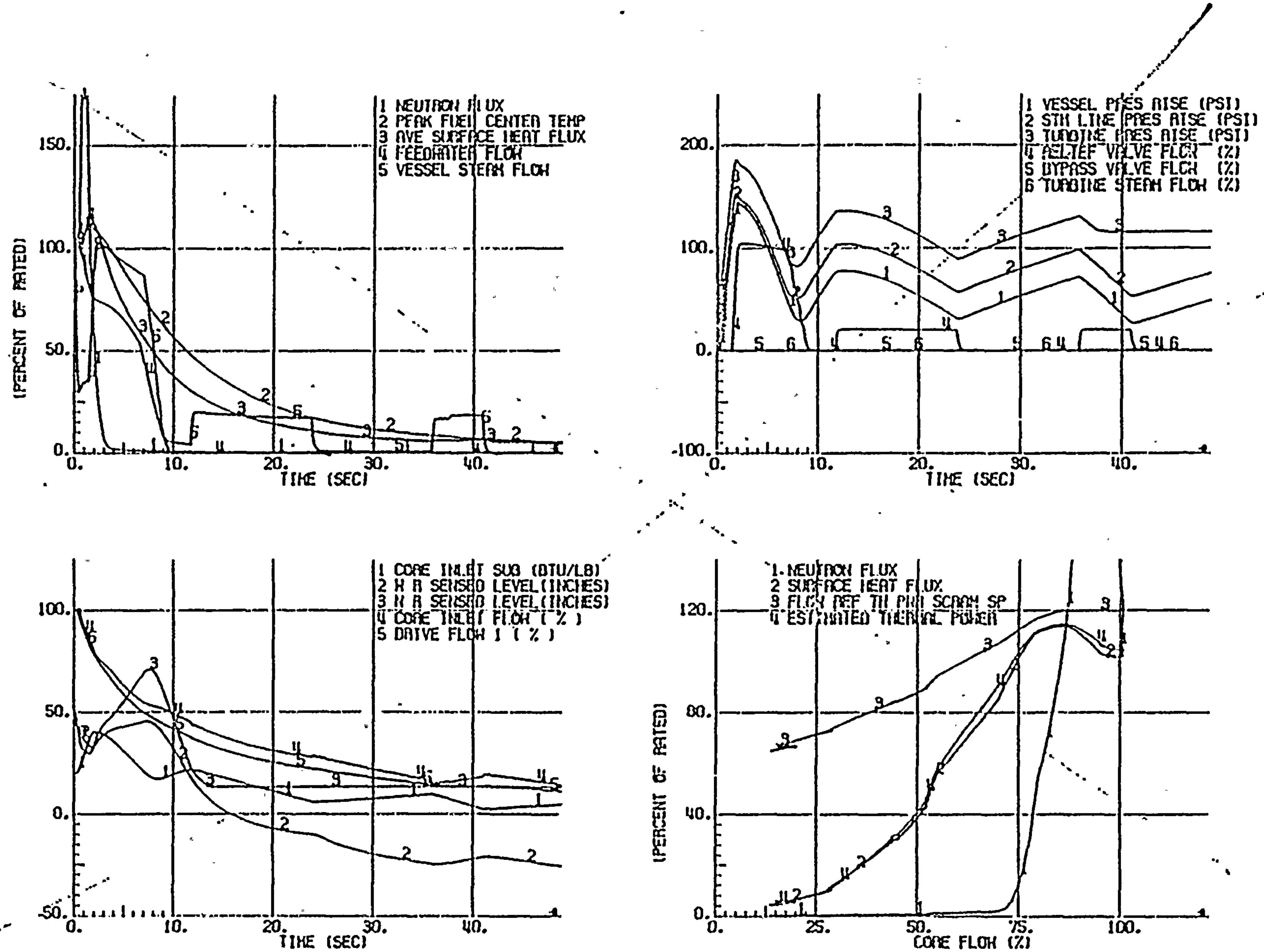
SEQUENCE OF EVENTS FOR FIGURE 15.2-4

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiates a recirculation pump trip (RPT).
0.10	Turbine stop valves closed.
~~0.14~~ 0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
~~1.33~~ 1.39	Group 1 relief valves actuated.
~~1.45~~ 1.52	Group 2 relief valves actuated.
~~1.55~~ 1.64	Group 3 relief valves actuated.
~~1.66~~ 1.77	Group 4 relief valves actuated.
~~1.78~~ 1.90	Group 5 relief valves actuated.
~~6.70~~	~~Feedwater turbines trip on L8 high water level.~~
~~7.1 (est.)~~	~~Group 5 relief valves start to close.~~
~~9.6 (est.)~~	~~All relief groups closed.~~

TABLE 15.2-4 - (Continued)

Time-sec	Event
~~11.73~~	~~Group 1 relief valves reactuated on high pressure~~.
~~24.6 (est.)~~	~~Relief group 1 closed~~.
~~35.79~~	~~Group 1 relief valves reactuated on high pressure~~.
~~36.01~~	~~Main steam line isolation and RCIC and HPCS systems initiation on low level (L2)~~.
~~41.7 (est.)~~	~~Relief group 1 closed~~.
~~50+~~	~~Group 1 relief valves cycle open and close on pressure~~

REPLACE WITH NEW FIGURE 15.2-27



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

TURBINE TRIP, TRIP SCRAM, BYPASS - OFF, RPT - ON

FIGURE 15.2-27

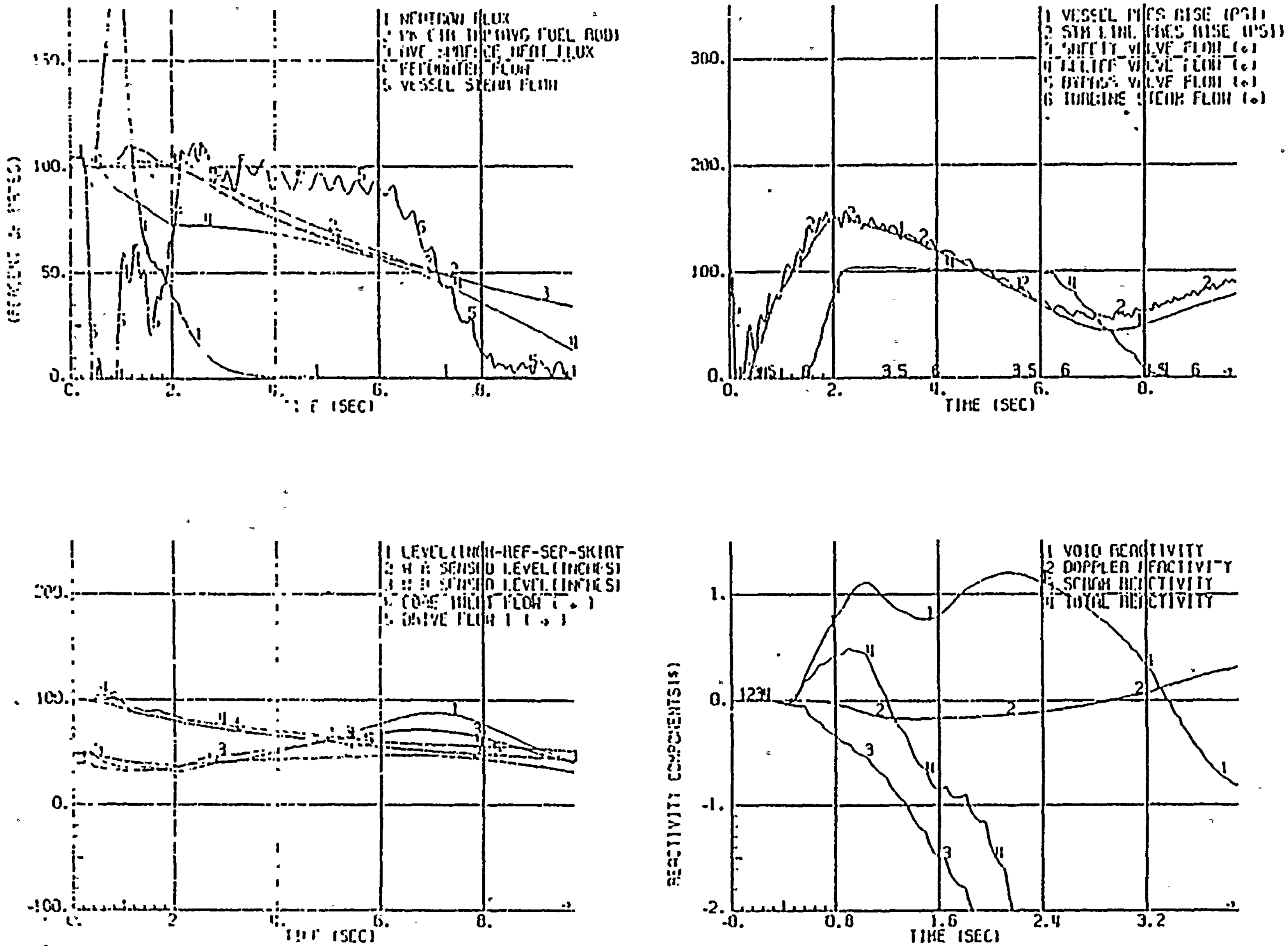


FIG. 15.2-4

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TURBINE TRIP WITHOUT BYPASS, TRIP SCRAM, RPT-ON.