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ENCLOSURES: Enc. 1 "Addl info re our 8-21-75 ltr re ATWS.....

Encl 2 "Corrections to the report entitled ATWS Study for the Monticello Plant" (NEDO-20846 Class 1 rev. 1, May 1975..."

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

November 10, 1975

Mr. D. L. Ziemann, Chief
Operating Reactors Branch # 2
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ziemann:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Response to August 21, 1975
ATWS Letter

This letter is in response to your August 21, 1975 letter regarding an Anticipated Transient Without Scram (ATWS) event at the Monticello facility.

The USAEC Technical Report "Anticipated Transients Without Scram for Water Cooled Power Reactors", WASH-1270, September, 1973 identified Monticello as a Class C plant stating that the need for backfitting for this class of plant should be considered on an individual case basis. Your August 21, 1975 letter stated that design modifications should be implemented at Monticello to reduce the probability or consequences of an ATWS event. As a Class C plant, the analyses required for Monticello by WASH-1270 did not treat ATWS as a new design requirement and therefore did not involve investigation of acceptable alternatives.

As a result of your August 21, 1975 letter, we have had discussions with all licensees of C plants which are similar to Monticello. A joint utility program is being formulated to evaluate ATWS alternatives for those plants. We expect to present to you in the near future a program, along with a schedule, designed to be compatible with the conditions of the February 28, 1975 Staff testimony on ATWS which states "...the probability of occurrence of an ATWS event with serious consequences is low enough to satisfy our safety objective today and for the next few years." (Docket No. 50-263, Supplemental Testimony of Nuclear Regulatory Commission Staff on Contention II-33, page 93.) We are prepared to work with you to resolve the appropriate backfit considerations on a schedule compatible with the safety objective stated in WASH-1270.

Your letter also requested additional information regarding the response of the Monticello plant to an ATWS event. Enclosure 1 to this letter provides

some of the additional information requested. Answers to other portions of your request should logically follow further ATWS analysis. The results of that study may show that certain of the requested information may be irrelevant or may need to be modified. Therefore, we will delay our response, where appropriate, until further evaluation of ATWS for C plants has developed the appropriate information.

A number of minor errors existed in our April 1, 1975 ATWS submittal which were identified to our NRC Project Manager in April and May. In order to establish a complete and correct record, we are including Enclosure 2 which provides corrected pages and instructions for inserting the new pages.

Yours very truly,

M. H. Voth /for

L. O. Mayer, PE
Manager, Nuclear Support Services

LOM/MHV/deb

cc: J. G. Keppler
G. Charnoff
MPCA
J. W. Ferman

Enclosures

Enclosure 1

The following information repeats the requests for additional information from the August 21, 1975 letter from D. L. Ziemann (USNRC) to L. O. Mayer (NSP) and provides the respective responses.

Request Number 1

Provide the peak torus water temperature reached during the MSIV closure ATWS. Provide and justify a torus water temperature limit. If the calculated temperature exceeds the limit, discuss the plant modifications needed to keep torus water temperature below the proposed limit. If the peak torus water temperature exceeds 170°F discuss plant modifications needed to keep this temperature below 170°F.

Response Number 1

Torus water temperature was calculated and is reported in responses 4,5 and 6 below. As stated in NEDO-20846, the justification of a torus temperature limit is part of a General Electric program currently underway.

Request Number 2

The analysis, as described in the Monticello ATWS report, takes credit for the operator initiating the standby liquid control (SLC) system five minutes after the ATWS event. Discuss the indications available to the operator to assure this manual initiation of the SLC.

Response Number 2

There are 5 aspects to be considered in answering this question. They will be considered individually in the order of increasing indication to the reactor operator.

A. ATWS event not involving reactor isolation.
B. ATWS event involving reactor isolation.
C. Scram is challenged; total lack of response.
D. Scram is challenged; partial response but no control rod movement.
E. Operator reaction to scram.

A. ATWS Events Not Involving Reactor Isolation - For purposes of responding to this question, ATWS events have been categorized into two groups. The events having the least impact on the plant are those not involving reactor isolation. If such an event occurred, the operator would observe the effects of such an event through changes in process variables which he continuously monitors in the control room (reactor power, pressure, system temperatures, radiation levels, etc.) He would also observe any change of state such as the automatic initiation of equipment. Significant deviations from steady state conditions are alarmed by the lighted and audible control room annunciator system which must be acknowledged by the operator to silence the alarm. The plant process computer monitors many of

the same parameters. When an alarm signal is received by the process computer, an audible alarm alerts the reactor operator and a hard copy of the alarm message is logged by the alarm typer.

There is widespread diversity among all of the functions, components and systems which provide the indications of an ATWS event and scram initiation.

B. ATWS Events Involving Reactor Isolation - The worst of ATWS isolation events is an MSIV closure as analyzed in report NEDO-20846 wherein it was assumed that the operator would respond by initiating the SLC system in 5 minutes. In addition to the general indications discussed above, the isolation events are characterized by the following indications:

1) Reactor pressure will increase rapidly with indications on a strip chart recorder before the reactor operator. Numerous alarms will light and sound almost instantaneously.

2) The neutron flux will spike upscale on the strip chart recorders before the reactor operator.

3) The relief valves will open which can be heard by operating personnel within the plant.

4) The torus pool temperature will be observed to increase.

Each of these parameters will be alarmed visually and audibly to the control room operator. This combination of events will immediately tell the operator that a scram should have accompanied this event and he will proceed with the procedure for a scram discussed below.

C. Scram is Challenged; Total Lack of Response - This situation can occur only in the unlikely event that a common mode failure affects a specific segment of the scram system. If the CMF affects sensors of a given function, the scram will be initiated by other process variables momentarily. For example, if an MSIV closure occurred along with sufficient failures of the position switches to prevent a scram, the reactor pressure sensors and the high neutron flux sensors would initiate a scram. It is difficult to postulate a CMF which would deprive the operator of the specific information that the scram system was challenged. Assuming, for the moment, that there is a total lack of information that the scram system was challenged, the reactor operator would still have the indications discussed in paragraphs A and B which would show the need to initiate the SLCS.

D. Scram Challenged; Partial Response but no Control Rod Movement - If the assumed common mode failure affected intermediate components between the sensors and the components implementing the scram, the more likely of the very unlikely hypothesized ATWS event, one might expect to have additional information available identifying the challenge of the scram system but with failure of rods to move. The scram will be annunciated directly to the operator and the plant process computer typers will begin printing a sequence of events log and a plant disturbance log. Immediately the operator's reaction will be to respond according to the scram procedure discussed below.

E. Operator Reaction to Scram - The first four steps of the procedure that an operator follows after a scram are as follows:

1) Announce over the plant paging system that a scram has occurred.
2) Place the reactor mode switch in the Shutdown position
3) Verify that all rods have been inserted by observing the digital position indication of each control rod displayed on the operator console.
4) Insert SRM and IRM detectors.

(Of these immediate steps it should be noted that by placing the reactor mode switch in the Shutdown position an automatic interlock acts to again initiate a scram. Also, when the SRM and IRM detectors are inserted and sense a high count rate within the core, they will initiate a scram.) After receiving all the indications discussed above, the operator will know that a scram should have occurred, and in accordance with step 3 of the scram procedure he will verify that the control rods have inserted properly. If he observes that none of the control rods have inserted, he will immediately initiate a manual scram. Because of the design of the scram system there is a possibility that should sufficient equipment fail to prevent the automatic scram, the manual scram will still function. In the unlikely event that the manual scram does not result in control rod movement, the operator would realize that SLC must be initiated to shut down the reactor. In the case of a failure to scram following an isolation event, sufficient information would be available to him within a few seconds upon which he would base his decision to initiate the SLC system.

The SLC system is initiated by actuating a single keylocked switch in the control room. No further operator action is required. The key to initiate this system is under the control of the shift supervisor. The shift supervisor reports to the control room immediately upon the announcement of a scram. It is extremely unlikely that he will be more than a minutes distance from the control room. The largest fraction of a shift supervisors time is spent in the control room or in an office adjoining the control room. It is therefore proper to assume that should an isolation event occur with a failure to scram, the operator would be made aware of the situation and have the capability to initiate the SLC system to correct the situation within five minutes.

Request Number 3

In figure 4-3 the relief valve flow oscillates between about 3,000 and 7,000 lb/sec from about 30 seconds to 95 seconds after the ATWS. At about 108 seconds the relief valve flow begins to oscillate between 3,000 and 14,000 lb/sec. Explain this difference in the peak relief valve flow.

Response Number 3

Please note that the ordinate of Figure 4-3 has been corrected to read "Flow Rate (lb/sec x 10^3)" rather than "Flow Rate (lb/sec x 10^4)."

Figure 4-3 shows the ATWS analysis assuming three plant modifications initiated upon high reactor pressure, recirculation pump trip, feedwater pump trip and ADS inhibit.

Upon MSIV closure, reactor pressure rises causing the recirculation and feedwater pumps to trip at 4 seconds. The former causes reduction in the flow through the core. Stoppage of feedwater causes the reactor water level to gradually drop (thereby further decreasing the core flow) and also reduces the sub-cooling of the core inlet flow. Both core flow decrease and core inlet sub-cooling decrease result in increased core average voids and therefore decreased core power. Therefore, after the initial pressure and power spikes subside (i.e., after about 30 seconds) the reactor power attains a level of approximately 30% of the initial valve. With MSIV's closed, this power is relieved from the reactor pressure vessel by steam flow through the relief valves which are assumed to operate in four groups. At this power level only two relief valve groups are sufficient to relieve all the energy generation. The opening and closing characteristics of the relief valves cause the relief flow to oscillate between approximately 350 lb/sec and 700 lb/sec (which indicates that the first group of relief valves is open and the second group is cycling).

The reactor level continues to drop due to continued power generation and lack of feedwater flow. When it reaches the low low level, the HPCI system is initiated. The HPCI flow starts at about 85 seconds and brings water of enthalpy 90 BTU/lbm into the reactor. This relatively cold water increases the core inlet sub-cooling resulting in slightly decreased core voids and increased core power. To relieve the increased power more relief valves are called upon to act. This combined with the dynamic characteristics of the relief valves causes the relief flow to oscillate between 350 lb/sec and 1,400 lb/sec after about 100 seconds (indicating that the first group of valves is open and the next three groups are cycling).

Requests Number 4, 5 and 6

The Technical Specifications present sodium pentaborate solution concentration versus net tank volume in Figure 3.4.1. The concentration varies from 10.8% to 21.4%. Perform the analysis using each of these concentrations. Justify the use of 13% as an initial condition listed in Table 3-1 of NEDO-20846. Also justify the poison reactivity worth and specify the reactor vessel volume.

In Section 4.4 of the Technical Specifications a minimum flow rate of 24 gpm for each of the standby liquid control system pumps is listed as a surveillance requirement. Perform the analysis using this value. In Table 3-2 of NEDO-20846 a 28 gpm flow rate per pump is listed. Provide your basis for using this value in your analysis. Specify the total volume of poison injected following the ATWS and indicate the required volume for both hot shutdown and cold shutdown.

It is stated that no accounting for possible non-homogeneous mixing was made since this would take a detailed evaluation. However, GE stated at a meeting with the staff on August 7, 1974, that tests were being conducted on borated water mixing phenomena. Demonstrate that your assumption of uniform mixing is consistent with the experimental data. Otherwise, perform a sensitivity study to show the effects of non-homogeneous mixing of the liquid poison, varying the mixing efficiency from 50% to 100%.

Responses Number 4, 5 and 6

A base case calculation for Monticello was provided in NEDO-20846; the inputs and assumptions are listed in that document. Sensitivity studies of those parameters identified in requests number 4, 5 and 6 are summarized in Table 1, below. An estimation of the effect of a lesser reactivity insertion rate on torus temperature and pressure is provided in Table 2 based on calculational results of the base case. This information, along with miscellaneous requested data presented in Table 3, can be used to assess the physical effects of parameters in question.

Please note that vessel pressure peak, fuel enthalpy peak and cladding oxidation are not affected by change in SLC reactivity insertion rate.

Additional information on this topic may be deemed appropriate for the generic study of ATWS for C plants.

Analytical studies treating the primary aspects of the mixing of the sodium pentaborate solution in the reactor vessel are underway at the present time. These studies are, at present, expected to be completed by the end of the first quarter of 1976.

TABLE 1

Reactivity Insertion Rates Corresponding to Conditions Other Than Those Used in the Base Case

Case	Condition as Different From Base Case	Corresponding Reactivity Insertion Rate (-¢/Sec) With 1 SLC Pump	With 2 SLC Pumps
1	None - (Base Case)	1.19	2.38
2	Sodium Pentaborate Concentration = 10.8%	0.9886	1.9772
3	Sodium Pentaborate Concentration = 21.4%	1.9589	3.9178
4	SLC Flow Rate = 24 gpm/pump	1.02	2.04
5	Mixing Efficiency = 50%	0.595	1.19
6	Mixing Efficiency = 75%	0.8925	1.7850

TABLE 2

Effect of SLC Reactivity Insertion Rate on the Peak Containment Pressure and Temperature (SLC Initiation Time = 5 Min.)

Reactivity Insertion Rate -¢/Sec.	Containment Peak Temperature °F	Containment Peak Pressure (psig)
2.38	184	5.1
1.19	213.9	6.9
0.595	274	48.6

TABLE 3

Miscellaneous Data Requested

Reactor vessel volume to normal water level	9,130 ft^3
Reactivity required to bring reactor from 100% power to hot shutdown	1.7%
Reactivity required to bring reactor from hot shutdown to cold shutdown	1.5%

Request Number 7

The staff has submitted to General Electric questions on NEDO-20626 (letter from V. Stello to I. Stuart, January 28, 1974, and letter from W. Butler to I. Stuart, April 9, 1975, copies are enclosed). Respond to the following questions as they apply to Monticello: 1, 4, 5, 6, 9, 12, 13, 16, 17, 310.1, 310.3, and 310.5.

Response Number 7

The requests for information referenced above were asked as part of the staff review of a generic study of Class B BWR plants. Each request will be reviewed in the appropriate perspective as part of the anticipated Class C generic program and addressed accordingly.

Request Number 8

Provide the bases for assuming thirty seconds for transport time of the sodium pentaborate solution from the storage tank to the vessel and for the liquid to become effective in the core.

Response Number 8

The thirty second transport time is made up of two segments. Approximately half of the time is required to pump the boron solution at rated flow to the sparger immediately below the core. The remaining 15 seconds is an estimate of the time it takes for the solution to mix with reactor water and become effective in reducing the reactivity of the core. Study of boron mixing discussed in the response to requests number 4, 5 and 6 will provide a better basis for transport time.

Enclosure 2

This enclosure transmits corrections to the report submitted by L. O. Mayer (NSP) to A. Giambusso (USNRC) on April 1, 1975 entitled "Anticipated Transients Without Scram: Study for the Monticello Generating Plant, NEDO-20846, March 1975." The changes include the following:

1. Either destroy the hard cover (which contains the same information as page i) or change the date from "March 1975" to "Revision 1, May 1975."

2. Destroy the pages of the original report listed below. Replace each page with the respective replacement page dated "May 1975" which is attached.

Superceded Pages

i
4
7
9
11
12
14
16
18

Regulatory Docket File

NEDO-20846
Class I
Rev. 1, May 1975

11-10-75

ANTICIPATED TRANSIENTS WITHOUT SCRAM

STUDY FOR THE MONTICELLO NUCLEAR GENERATING PLANT

BOILING WATER REACTOR PROJECTS DEPARTMENT • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

2. ANALYSIS GUIDES AND RESULTS

2.1 REACTOR COOLANT SYSTEM PRESSURE

WASH 1270 requires comparison of primary stress to that of the emergency conditions of the ASME Nuclear Power Plant Component Code, Section III. On consideration of this guide and examination of the system, the WASH 1270 guide translates to a vessel pressure of 1500 psig. NEDO-10349 uses 2700 psig as the vessel pressure that can be accommodated without structural failure.

2.2 FUEL THERMAL AND HYDRAULIC PERFORMANCE

WASH 1270 requires evaluation of any fuel cladding degradation or significant fuel melting. These subjects, including pertinent failure mechanisms, were discussed at length in NEDO-10349, Section 5.1.3. The application of the guide does not change from that made in the previous report. With respect to prompt failures, an energy deposition guide of 280 cal/gm has been selected. It has been shown that fragmentation is avoided at oxidation levels of less than 17% by volume.

2.3 CONTAINMENT CONDITIONS

WASH 1270 requires comparison of containment pressure to the design pressure. The containment design pressure for the Monticello Plant is 56 psig. NEDO-10349 uses the membrane yield limit of the primary containment which is 108 psig, as a guide.

2.4 SUMMARY OF RESULTS

Table 2-1 summarizes the results of the analyses.

Table 2-1
SUMMARY OF RESULTS
MONTICELLO NUCLEAR GENERATING PLANT

MSIV CLOSURE WITH FAILURE TO SCRAM

Parameter	Bounding Value by Analysis with RPT, Feedwater System Modification, and 5 minute SLC Initiated
Vessel Pressure (psig)	1307
Fuel Enthalpy (cal/gm)	<150
Cladding Oxidation (%)	<1
Containment Pressure (psig)	6.9

3.3 EQUIPMENT CHARACTERISTICS

The characteristics of the important pieces of equipment used to mitigate the consequences of failure to scram are listed in Table 3-2.

Table 3-2
EQUIPMENT PERFORMANCE CHARACTERISTICS

Parameter	Characteristic
Relief Valve System Capacity (% NBR Rated Steamflow)	74.4
Relief Valve Setpoint Range (psig)	1080 +1%
Relief Valve Time Delay (sec)	0.4
Relief Valve Opening Time (sec)	0.1
Control Liquid Injection Rate per Pump (gpm)	28
Delay Time from Control Liquid Initiation to begin Shutdown (sec)	30
HPCI Flow Rate (lb/sec)	415
RHR HX Effectiveness (Btu/sec-°F)	200
Recirc Pump Trip Reactor Pressure or Water Level Sensor and Logic Time Delay (sec)	0.53
Recirc Pump System Inertia Constant (sec)	5

The long-term effects of an ATWS event depend in part on initiation of additional equipment to mitigate the consequences. The initiation times for which credit is taken in the various analyses are:

Standby Liquid Control System:	Initiation — 5 minutes after the event
RHR System:	Initiation — 10 minutes after the event

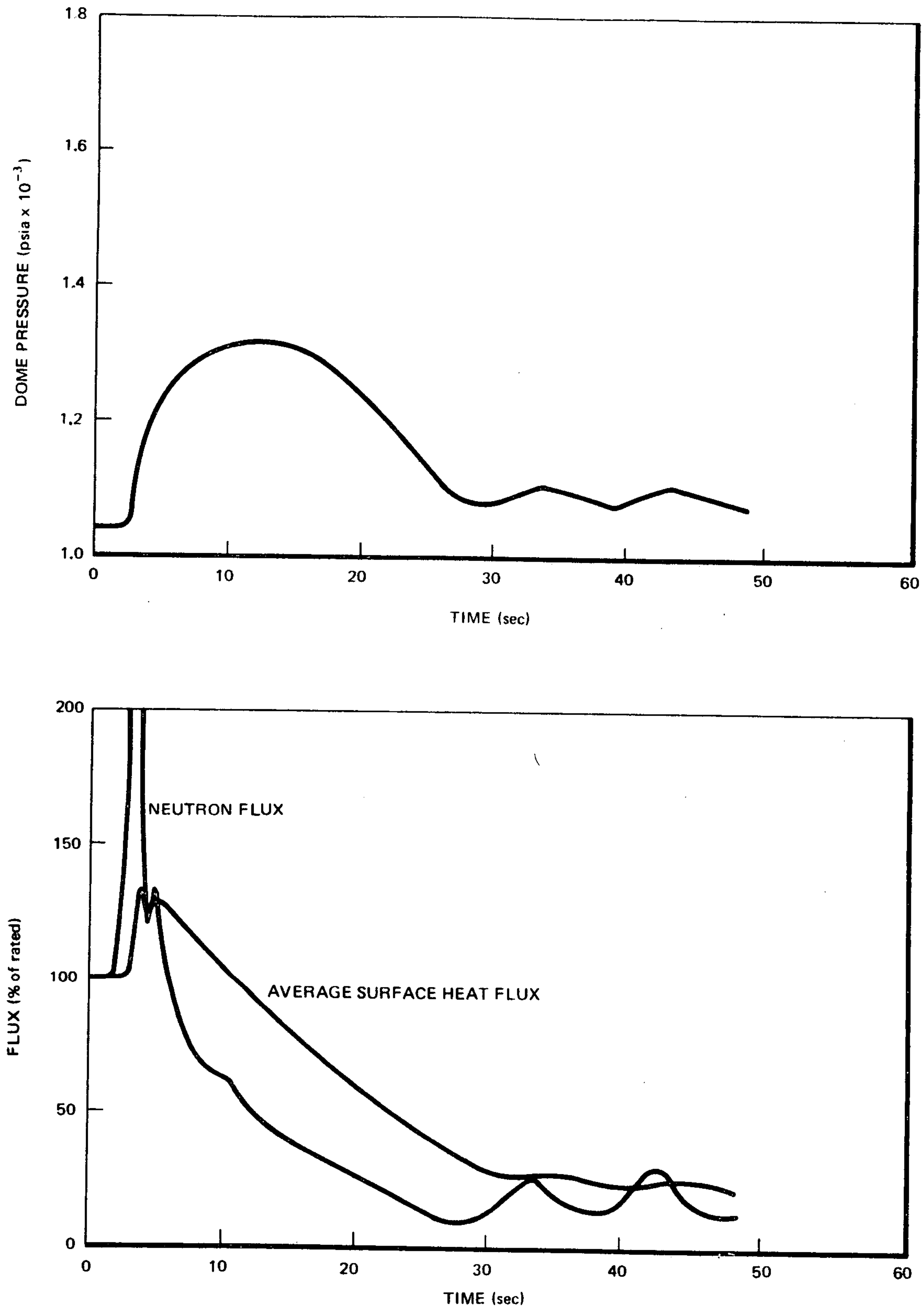

Figure 4-1. Monticello MSIV Closure Transient – ATWS Response with ATWS Modifications

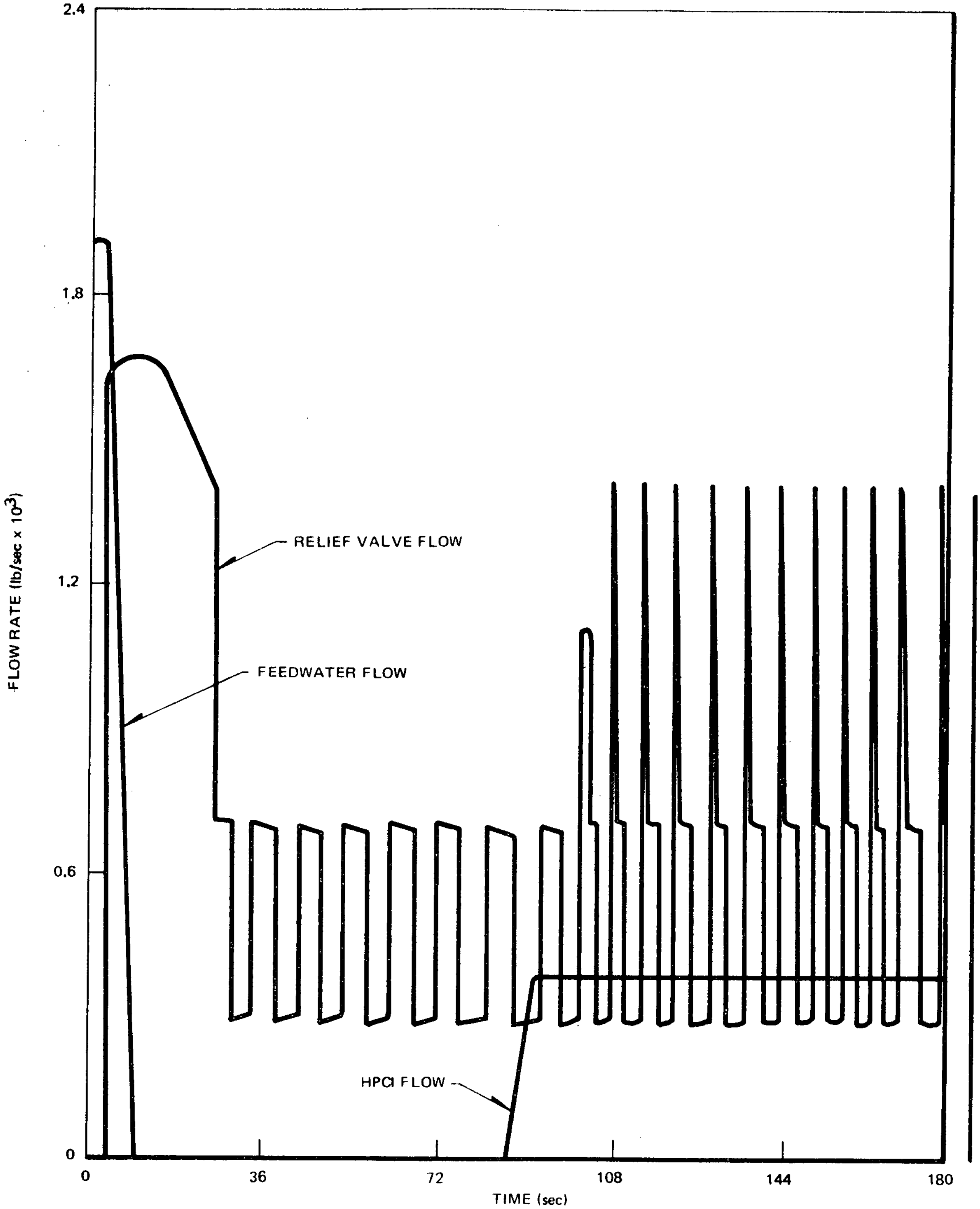

Figure 4-3. Monticello MSIV Closure Transient – ATWS Response with ATWS Modifications

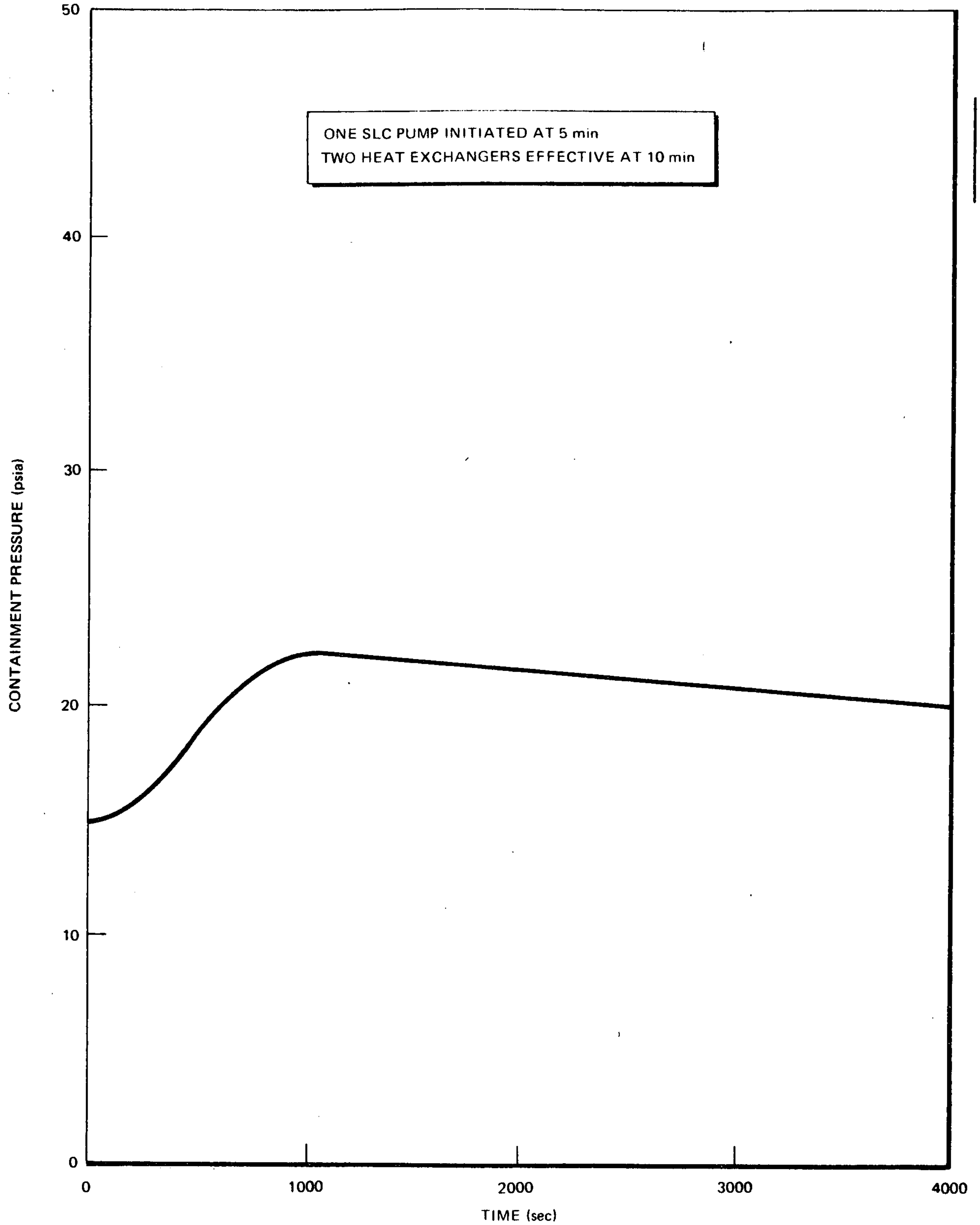

Figure 4-4. Monticello MSIV Closure Transient – ATWS Containment Response with ATWS Modifications

4.3.3 Containment

For purposes of this report, the term containment will be used to include the drywell as well as all those enclosed spaces which are affected by the steam released to the suppression pool. Response refers to the action of the pressure and temperature in the containment as steam is released to the suppression pool and drywell.

All steam passes through the relief valves and enters the suppression pool. All steam that enters the pool is assumed to be condensed and the pool temperature is effected accordingly. Both RHR heat exchangers are assumed to be actuated at 10 minutes, but the energy from the steam release exceeds their heat removal capacity at the initial temperature so that pool temperature continues to increase until the RHR heat exchangers capacity is equal to the energy being generated by decay heat.

Containment pressure will also increase along with pool temperature. The containment pressure transient is shown in Figure 4-4. The maximum containment pressure is 7.3 psig, which is within the guide values.

4.4 COMPARISON TO WASH 1270

Appendix A, paragraph II.C.1 of WASH 1270 requests comparison of three functions to specified analytical guides. Table 4-1 provides a comparison of the analytical results with the WASH 1270 and General Electric guides.

Table 4-1
SUMMARY OF RESULTS
MONTICELLO NUCLEAR GENERATING PLANT

MSIV CLOSURE WITH FAILURE TO SCRAM

Functional Comparison Parameter	WASH 1270 Comparison Value	General Electric Suggested Guide	Bounding Value by Analysis with RPT, Feedwater System Modification, and 5-min. SLC Initiation
Vessel Pressure (psig)	1500	2700	1307
Fuel Enthalpy (cal/gm)	280	280	<150
Cladding Oxidation (%)	17	17	<1
Containment Pressure (psig)	56	108	6.9

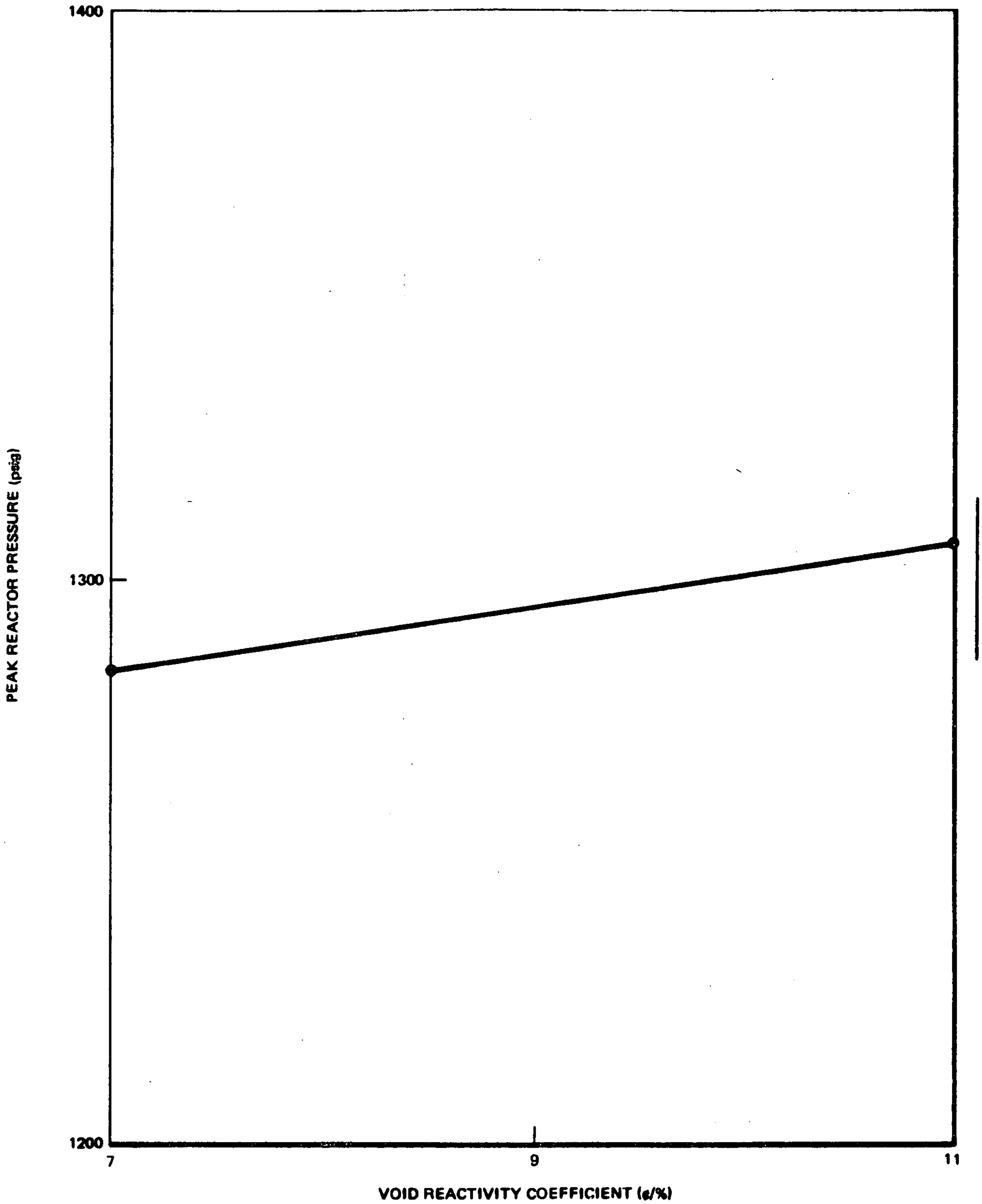

Figure A-1. Monticello ATWS Response with ATWS Modifications – MSIV Closure Transient

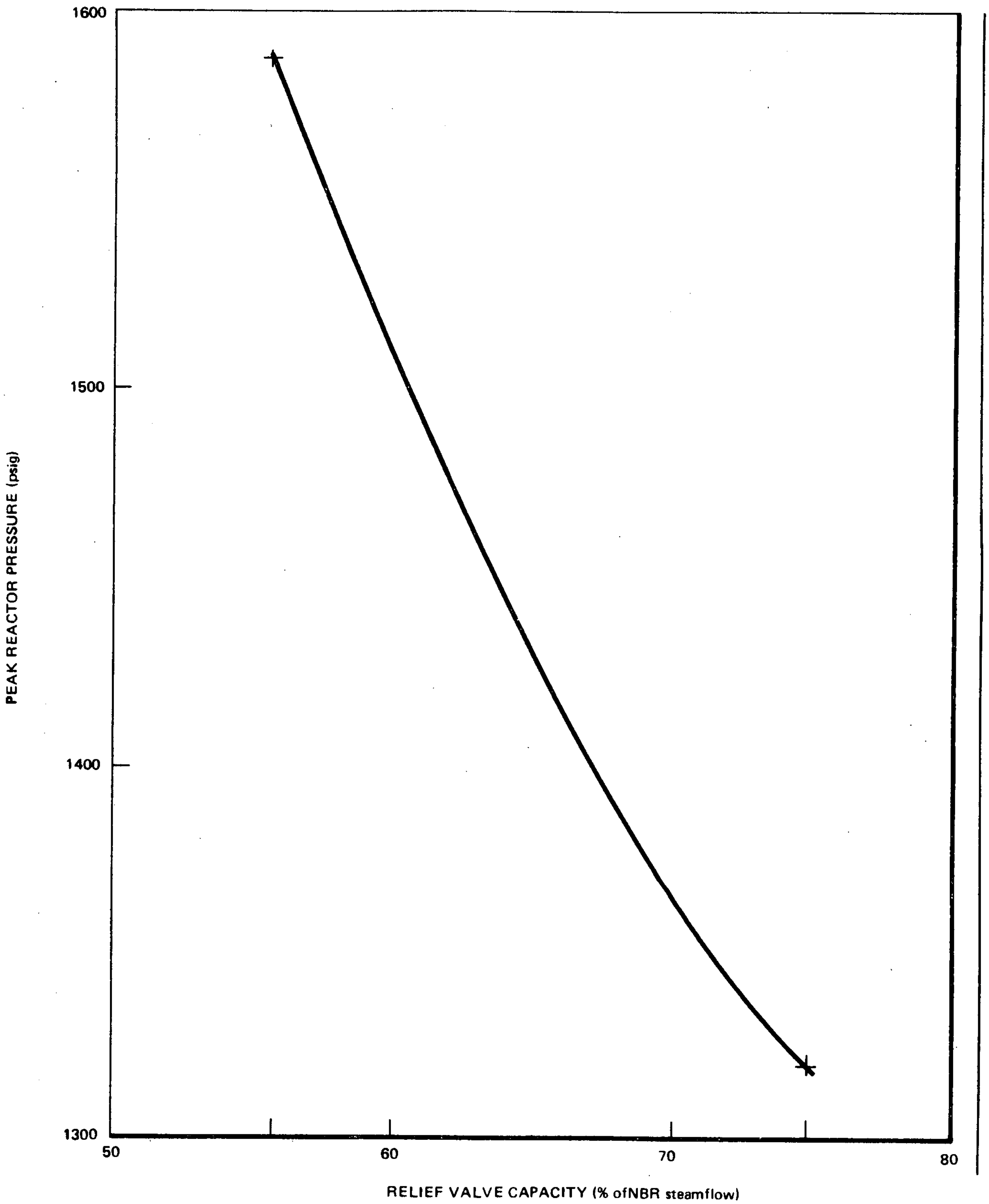

Figure A-3. Monticello ATWS Response with ATWS Modifications – MSIV Closure Transient