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USE ONLY

ANTICIPATED TRANSIENTS WITHOUT SCRAM STUDY FOR BIG ROCK POINT POWER PLANT

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1. INTRODUCTION

1.1 SUBJECT OF REPORT

In September 1973, the NRC issued their Technical Report on Anticipated Transients Without Scram (ATWS) for Water Cooled Reactors (WASH-1270). That report defined three categories of plants. The first or A category requires improved reactor shutdown systems and is applicable to plants for which construction permit applications are made after October 1, 1976. The second or B category requires provision to make the consequences of ATWS acceptable and is applicable for plants for which the need for provision for ATWS is noted in the NRC Safety Evaluation Report or the ACRS Report and for plants for which the construction permit applications are or have been made prior to October 1, 1976, and NRC SER is not yet issued. The remainder of the plants fall into the third or C category which requires an analysis of ATWS consequences so that the NRC staff can evaluate whether there is need for plant changes to resolve the ATWS issue.

The Big Rock Point Power Plant falls in the C category as defined in the Appendix B of WASH-1270. This report describes the studies performed for the Consumers Power Company in response to the requirement of an ATWS consequences transient analysis for the Big Rock Point Plant.

1.2 PREVIOUS STUDIES

The subject of transient evaluation without scram protection has been an NRC concern for some time. This comes out of a desire to better understand and protect against common mode types of failures (CMF). General Electric has responded to this concern with topical reports NEDO-10189, entitled "An Analysis of Common Mode Failures in GE BWR Protection and Control Instrumentation," July 1970, and NEDO-10349, "Analysis of Anticipated Transients Without Scram," March 1971. In addition, General Electric has issued NEDO-20626, "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram," October 1974, which responds to the B category of BWR WASH-1270 requirements for an ATWS mitigation design.

Both NEDO-20626 and this report are based on the premise that a failure to scram occurs coincident with an abnormal operating transient. This premise is assumed in this report to satisfy the requirement of WASH-1270 for analysis of the consequences of anticipated transients in the event of a postulated failure to scram. WASH-1270 states that the likelihood of a severe ATWS is considered to be acceptably small in view of the limited number of plants now in operation, the reliability of the current protective system designs, and the expected occurrence rate of anticipated transients of potential safety significance. WASH-1270 thus does not justify the imposition of any new requirement for hardware changes on existing plants. The General Electric Company is in complete agreement with this position. This report should not, therefore, be construed as a recommendation for plant modifications for the purpose of mitigating the consequences of a postulated ATWS event. It is an exploratory engineering study to evaluate ATWS as required by WASH-1270.

1.3 APPROACH USED IN THIS STUDY

This report provides a representative treatment of the main aspects of WASH-1270 requirements. Since the primary aspects of NEDO-10349 have been completely reconfirmed by further calculations, it was used as a guide for identification of the most limiting type of transients relative to each of the analysis guides when failure to scram was considered.

Attention was directed toward the transients which have the highest probability of occurrence. This is consistent with the stated objective of WASH-1270. Very infrequent events are not considered as they make no significant contribution to public safety considerations when combined with the low probability of failure to scram.

Previous analyses have shown that among the events reasonably expected to occur, the ones which cause the most severe ATWS conditions are those which initiate fast shutoff of the steam flow from the reactor, such as turbine-generator trip or the closure of the main steam isolation valves. For the Big Rock Point plant the MSIV closure time is much slower than in the later plants. This would make the MSIV closure event relatively much less severe. Also, the large bypass system on the

Big Rock Point plant makes the turbine-generator trips relatively mild. The analysis given in this report, in which a turbine-generator load rejection is assumed to cause reactor isolation by taking no credit for the action of the bypass valves, constitutes an extremely conservative bounding case.

In analysis, ATWS behavior can be separated into the behavior of the nuclear boiler and that of the containment. In this report the nuclear boiler dynamics is not calculated since previous studies done for FHSR input, APED-4093 (Reference 1), contain safety valve sizing analyses which are pertinent to the present work. Specifically they analyze the transient of turbine load rejection with safety valve operation, without bypass and without scram. This analysis, therefore, represents a bounding ATWS event. Comparison of reactor peak pressure and fuel response with WASH-1270 criteria shown in this report are based on these safety valve sizing analyses.

The main concern here is to calculate the response of the containment in an ATWS event. As far as the containment response is concerned, the difference between different reactor isolation events without scram is insignificant when the initial power level is the same. As the reactor response for an ATWS event of isolation caused by load rejection without bypass is available from Reference 1, containment response is studied for this transient.

2. SUMMARY OF RESULTS

Table 2-1
WORST REACTOR ISOLATION WITHOUT SCRAM
(BOUNDED BY TURBINE LOAD REJECTION NEGLECTING BYPASS)

Functional Comparison Parameter	General Electric Guide	Value Analysis
Reactor Vessel Pressure (psig)	2700	1587[a]
Fuel Enthalpy (cal/gm)	280	<165
Cladding Oxidation (%)	17	<1
Containment Pressure (psig)	54	57.7

[a] Peak reactor pressure increase taken from Reference 1, Figure 15, Run 3.

3. ANALYTICAL BASES

3.1 ANALYTICAL METHODS

The first step in the analysis of ATWS behavior is to calculate the dynamics of the nuclear boiler. This calculation is already bounded by the safety valve sizing analysis reported in APED-4093. That analysis used an analog model of the reactor and the details are documented there.

The response of the containment is evaluated using the techniques described in the General Electric Licensing Topical Report NEDO-20533, dated June 1974, entitled "The General Electric Mark III Pressure Suppression Containment System Analytical Model." The engineering methods used in this report are appropriate for use in all containment designs. The specific methods used are described in Section 3 of NEDO-20533. The following assumptions were used in analyzing the containment response to an ATWS event.

1. The steam release from the reactor to the containment sphere as a function of time is taken from the safety valve sizing analysis reported in Reference 1.

2. The steam entering the containment sphere has an enthalpy of 1190 Btu/lbm.

3. The air-water vapor system (excluding the containment spray water) is in thermodynamic equilibrium. The equilibrium condition is dependent on the initial conditions and the net steam and enthalpy dump into the containment.

4. No heat is lost to the outside by heat transfer through the containment wall. However, the cooling effect by heat transfer into the containment wall is taken into account. The containment wall is assumed to be at a single temperature and heat transfer between the air-water vapor system inside the containment and its wall is assumed to take place until thermal equilibrium between the two is achieved. Similar cooling effect of other structures in the containment is neglected.

5. The heat transfer coefficient between the containment air-water vapor system and the containment wall is assumed to be a function of the air-to-water-vapor weight ratio inside the containment.

6. The effect of the containment spray is limited to abstraction of heat from the air-water vapor system.

7. The effectiveness of the containment spray is assumed to be 70%. (The effectiveness is defined as the ratio of the temperature rise in the spray water to the difference between the containment air-water vapor system temperature and the initial temperature of the spray.)

Effects of the fuel surface heat transfer are estimated using the "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application" code as described in NEDO-10958, dated November 1973. Further evaluations of fuel and cladding temperatures are derived using methods basic to the LOCA analysis also described in NEDO-10958. However, in this calculation the nuclear power generation and fuel heat flux data needed as input are taken from the safety valve sizing analysis of Reference 1.

Other areas which are not rigorously treated in the analytical bases of this report include nonhomogeneous mixing of liquid poison.

3.2 INITIAL OPERATING CONDITIONS

Table 3-1 lists the initial conditions for the most important plant parameters. These are chosen as representative of the conditions of the reactor at the beginning of an event for which ATWS impact is evaluated.

Table 3-1
INITIAL CONDITIONS

Parameter	Initial Condition
Reactor Operating Pressure (psig)	1335[a]
Power (MWt)	240[b]
Steam/Feed Flow (lbm/sec)	278[c]
Containment Sphere Free Volume (ft³)	940,000[b]
Containment Sphere Thickness (in.)	0.702 (min)[a]
	0.875 (max)[a]
Initial Pressure of Containment (psig)	0.0[d]
Initial Temperature of Containment (°F)	100[d]
Relative Humidity (%)	100[d]

[a] From Reference 4
[b] From Reference 2
[c] Based on 240 MWt power and feedwater enthalpy of 347.1 Btu/lbm
[d] From Reference 3

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
Safety Valves	
Number	6[a]
Capacity (lbm/sec/valve)	63.9[a]
Setpoint Range (psig)	1535-1585[a]
Emergency Condenser	
Number of Tube Bundles	2[b]
Capacity (Btu/hr/bundle)	16×10^6[b]
Initiation Signal (Reactor Pressure Rise, psig)	110[b]
Delay Time (sec)	4[c]
Containment Spray	
Number of Sets	2[b]
Flow Capacity (gpm/set)	400[b]
Initial Temperature (°F)	80[c]
Effectiveness (%)	70[c]
Delay Time from Liquid Control Start to Beginning of Shutdown (sec)	30[c]
Time Required to Complete the Injection of Control Liquid (sec)	300[b]

Table 3-2
EQUIPMENT PERFORMANCE CHARACTERISTICS (Continued)

Parameter	Characteristic
Initiation Time	
Standby Liquid Control (sec)	300[c]
Containment Spray	300[c]
Containment Spray Delay	30[c]
Reactor Recirculation Pump Trip	None[e]

[a] From Reference 4
[b] From Reference 2
[c] Assumed
[d] From Reference 3
[e] No recirculation pump trip is assumed to conservatively evaluate the calculated effects of the ATWS in the analysis. Appendix A calculates the effect of assuming recirculation pump trip initiated at various time instants.

4. ANALYSIS GUIDES

4.1 GENERAL

WASH-1270 asks for comparison of the ATWS transient to show whether:

a. Calculated reactor coolant system transient pressure exceeds a value such that the maximum primary stress in the system boundary is equal to that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III, or the

b. Effects of the ATWS event result in significant fuel cladding degradation or significant fuel melting, or the

c. Calculated containment pressure exceeds the design pressure of the containment structure.

Since these guides are applied to all reactor types, and consequently are rather general, it is necessary to interpret the guides with respect to the Big Rock Point BWR design. The guide interpretations are discussed in the following paragraphs and compared to the guides proposed in NEDO-10349. General Electric feels that, for events as improbable as those associated with failure to scram, the limiting criteria in NEDO-10349 are sufficient for maintenance of public safety.

4.2 REACTOR COOLANT SYSTEM PRESSURE

4.2.1 Reactor Coolant System Boundary Pressure

On consideration of this guide and examination of the system, the WASH-1270 guide translates to a vessel pressure of 2040 psig. NEDO-10349 recommends 2700 psig as the vessel pressure that can be accommodated without structural failure.

4.3 FUEL THERMAL AND HYDRAULIC PERFORMANCE

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.

4.4 CONTAINMENT CONDITIONS

The containment sphere design pressure of the Big Rock Point Plant is 27 psig. NEDO-10349 recommends the membrane yield limit of the primary containment which would be 54 psig.

5. DEFINITION OF EVENTS

The occurrence of a common mode failure which completely disables the reactor scram function is a very low probability event. Therefore, no significant risk to public safety is presented by the combination of an infrequent event and a common-mode failure which prevents scram. Thus, attention is focused on those transient situations which have a relatively high expected frequency of occurrence at a power condition at which serious disturbance might result.

BWR analyses separate reactor transient duty into three main areas: normal plant maneuvers, abnormal transients, and design basis accidents. The middle group spans a very wide range of occurrence, from more than once per year to less than once per 40 years. Other nuclear reactor suppliers have for some time been defining this diverse group of abnormal transients into two categories of more- and less-frequent events. (ANSI-N18.2, Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants). All vessels and related components are also designed with strong considerations for the frequency as well as the magnitude of the thermal cycles caused by the transients. Operational experience from the growing list of plants on line provides valuable input to the understanding and establishment of reasonable rates of occurrence for events of importance in this study.

The following transients, derived for ATWS consideration on the basis of operational experience, have the potential of a frequency of occurrence of at least once in 4 years of reactor operation at power conditions such that a significant transient results and scram is called upon to shut down the reactor.

1. Turbine/generator trip
2. Closure of all main steam isolation valves
3. Pressure regulator failure in open direction
4. Feedwater controller failure to maximum demand
5. Loss of feedwater flow
6. Loss of auxiliary power.

Further descriptions of these events are contained in NEDO-10349. These events cover the spectrum of transients which might require shutdown and are representative of the experience record of operating BWR's.

The results of previous analyses reported in NEDO-10349 and NEDO-20626 show that the most severe transient in the above group of ATWS events is the closure of the MSIV. These two reports mainly address BWR plants constructed subsequent to Big Rock Point. For Big Rock Point the MSIV closure time is relatively much longer. The worst ATWS event for Big Rock Point would then be the reactor isolation caused by turbine-generator trip when bypass valve action is neglected. It is this event that is presented as the bounding analysis in this report. The analyses of NEDO-10349 and NEDO-20626 further show that as far as the dump of scram mass and energy into the containment is concerned, reactor isolation ATWS events caused by turbine trip neglecting bypass and MSIV closure are nearly the same. As steam release to containment data is already available from previous analyses (Reference 1) for the former ATWS event and as our main additional concern here is the containment response, containment response analyzed here is for this bounding case. However, it should be noted that the expected frequency of occurrence of the transient of load rejection with bypass failure is far below the value of once per four reactor years that is considered significant from the ATWS risk viewpoint. Full load rejection with normal bypass operation is not expected to call for a reactor scram.

Previous ATWS analyses of other BWR plants have shown that recirculation pump trip considerably reduces the severity of ATWS transients. The present analysis of the bounding case neglects this beneficial effect of recirculation pump trip which is the usual operator action on loss of turbine load. (Parametric studies of the effects of the recirculation pump trip are given in Appendix A.) For these reasons the turbine trip transient, even in the event of a postulated failure to scram, will be much less severe than the calculations presented here.

6. ANALYTICAL RESULTS

6.1 DESCRIPTION OF THE WORST ISOLATION EVENT (LOAD REJECTION NEGLECTING BYPASS)

6.1.1 Overview of Response Without Scram

The description of this event given here is based on the present analysis and that of Reference 1. The reactor behavior can be separated into a short-term condition — involving reactor pressure rise and fuel response (0-48 seconds) — and a longer-term condition (to several hundred seconds) — involving coolant and containment conditions — until the reactor is shut down.

As the turbine valve closes, due to the assumed lack of bypass, the reactor pressure rises. This causes the collapse of the voids in the reactor core resulting in a nuclear power spike. The pressure rise is limited by the safety valve operation. The reactor then assumes a new state (at a slightly higher thermal power) dictated by the safety valve setpoint and capacity. The safety valve steam flow released into the containment sphere pressurizes it.

In the absence of the scram function the long-term aspects of the ATWS event are terminated by inserting negative reactivity into the core by means of the standby liquid control system. The analysis shows the plant response for the case in which the liquid control is initiated at 5 minutes. As the steam flow to the containment sphere is terminated by reactor shutdown, the containment spray system recools the sphere, thereby bringing down its temperature and pressure. The emergency condenser which is initiated by high reactor pressure, reduces the steam output into the containment sphere. The metal containment wall absorbs heat from the air-water vapor mixture, thereby reducing the severity of the containment pressure and temperature conditions.

Present operating procedures at Big Rock Point call for the immediate trip of a reactor recirculation pump on certain plant transients. In the case of turbine trip without scram and without bypass, such a trip of a recirculation pump would reduce core flow which increases the core voids. This reduces the power and steam output of the reactor and reduces the severity of the ATWS event, especially in the containment sphere. The analysis given here neglects the beneficial effects of operator-initiated recirculation pump trip. The results presented here are therefore conservative. The effect of tripping the recirculation pumps at various times after the occurrence of an ATWS transient is given in Appendix A.

6.1.2 Sequence of Events for the Worst ATWS Isolation Event (Load Rejection Transient Neglecting Bypass)

	Event	Time (sec)
1.	Stop Valve Trips	0
2.	Reactor Pressure Begins to Rise	0
3.	Emergency Condenser Initiated	~3*
4.	Safety Valves Open	~3**
	Operator-Initiated Recirculation Pump Trip	
5.	Liquid Control Reaches Core	330
6.	Containment Sprays Begin Operation	330
7.	Hot Shutdown is Achieved	630
8.	Containment Temperature and Pressure Peak	~600

*Flow begins after 4-second delay.

**Neglected in the analysis for conservatism in the calculated effects.

6.2 SHORT-TERM CONDITIONS

6.2.1 Primary System Pressure

As stated before, the reactor pressure response is covered by the earlier safety valve sizing analyses of Reference 1 (Figure 15). Of the five different runs shown in Figure 15 of Reference 1, the one numbered 3 is for a setpoint of 200 psi above the operating pressure for the first safety valve. This value is the same in the present case (see Tables 3-1 and 3-2), and therefore Run 3 of Figure 15, Reference 1, is most appropriate for the present discussion. However, the capacity of the safety valves used in Reference 1 (Figure 15) was 200% of rated reactor flow. In the present case it is (63.9 x 6)/278 = 138%. This difference must be kept in mind in the present discussion of reactor pressure response. Taking the peak pressure rise of 252 psi from Reference 1, Figure 15, Run 3, the peak reactor pressure in the present ATWS discussion would be (1335 + 252) = 1587 psig, which is well below the design pressure of 1700 psig, and therefore is clearly acceptable.

6.2.2 Power and Fuel Response

An estimate was made for the maximum cladding oxidation and peak enthalpy experienced by the fuel for the worst reactor isolation case of full load rejection without bypass. The neutron flux (fission power) and fuel heat flux transients used in this analysis are shown in Figure 1. These are again taken from Reference 1, Figure 15, Run 3, with the fuel heat flux being assumed to be proportional to the fuel center temperature. The results obtained for the fuel response are similar to those in NEDO-20626 for the BWR/4. In this case the value of the cladding oxidation is far less than 17% by volume (<1%) and the peak fuel enthalpy is less than 280 cal/gm (<165 cal/gm). These values clearly demonstrate satisfactory fuel performance.

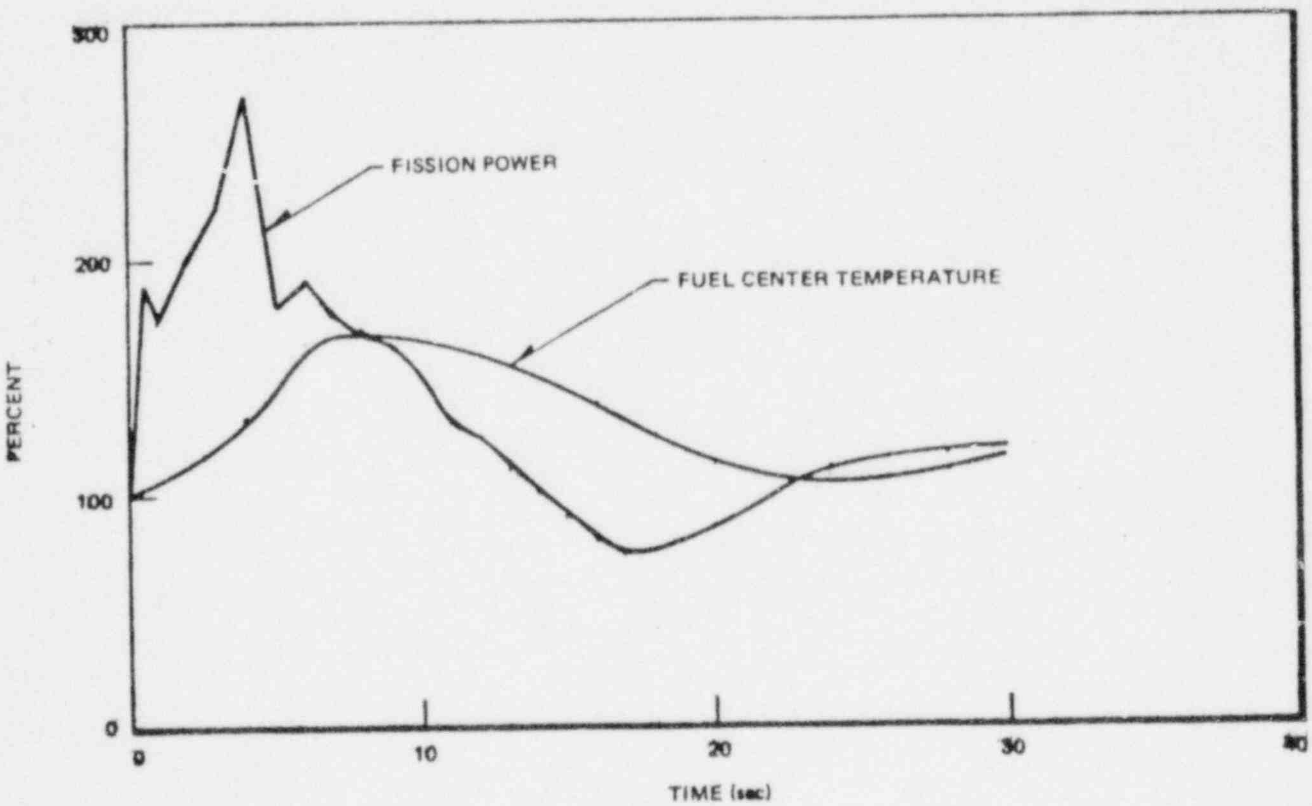

Figure 1. Fission Power and Fuel Center Temperature Transients Used in the Fuel Response Estimate in an ATWS Load Rejection Without Bypass. Data Taken From the Safety Valve Sizing Analysis of Reference 1, Figure 15, Run 3.

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6.3 LONG-TERM CONDITIONS

6.3.1 Power Shutdown

Since WASH 1270 requires the assumption of a simultaneous common-mode failure of all control rod drive mechanisms, no credit for normal or emergency control rod motion can be taken in the transient analysis. Consequently, another method must be used to bring the reactor core to shutdown. The alternative method available at the Big Rock Point plant is the use of the standby liquid control system which injects a sodium pentaborate solution into the reactor. For the purpose of analysis, the shutdown using the standby liquid control system was assumed to occur as follows: The operator has 5 minutes to evaluate the situation and decide to initiate the injection of the liquid poison. Thirty seconds were allowed for the transport time of the liquid from the storage tank to the vessel and to become effective in the core. Negative reactivity was then assumed to be inserted linearly until hot shutdown is achieved.

Having brought the reactor to the hot shutdown condition, time is now available for the operator to determine what is to be done next. If the postulated ATWS event has really occurred, he must take the necessary action to bring the plant to cold shutdown.

6.3.2 Containment

For the purposes of this report, the term containment is used for all the enclosed spaces affected by steam release into the containment sphere. Containment pressure and temperature refer to the condition of the air-water-vapor system inside the containment sphere. Containment wall temperature refers to the temperature of the metal content of the containment wall in its role as a heat source or sink with respect to thermal interaction with the air-water-vapor system inside. Containment response refers to the behavior of the pressure and temperature of the air-water-vapor mixture inside the containment sphere.

The reactor vessel steam flow used for the calculation of containment response is shown in Figures 2 and 3. Figure 2 is plotted from Reference 1, Figure 15, Run 3. Here the vessel steam flow from 0 to 40 seconds is shown. The flow here

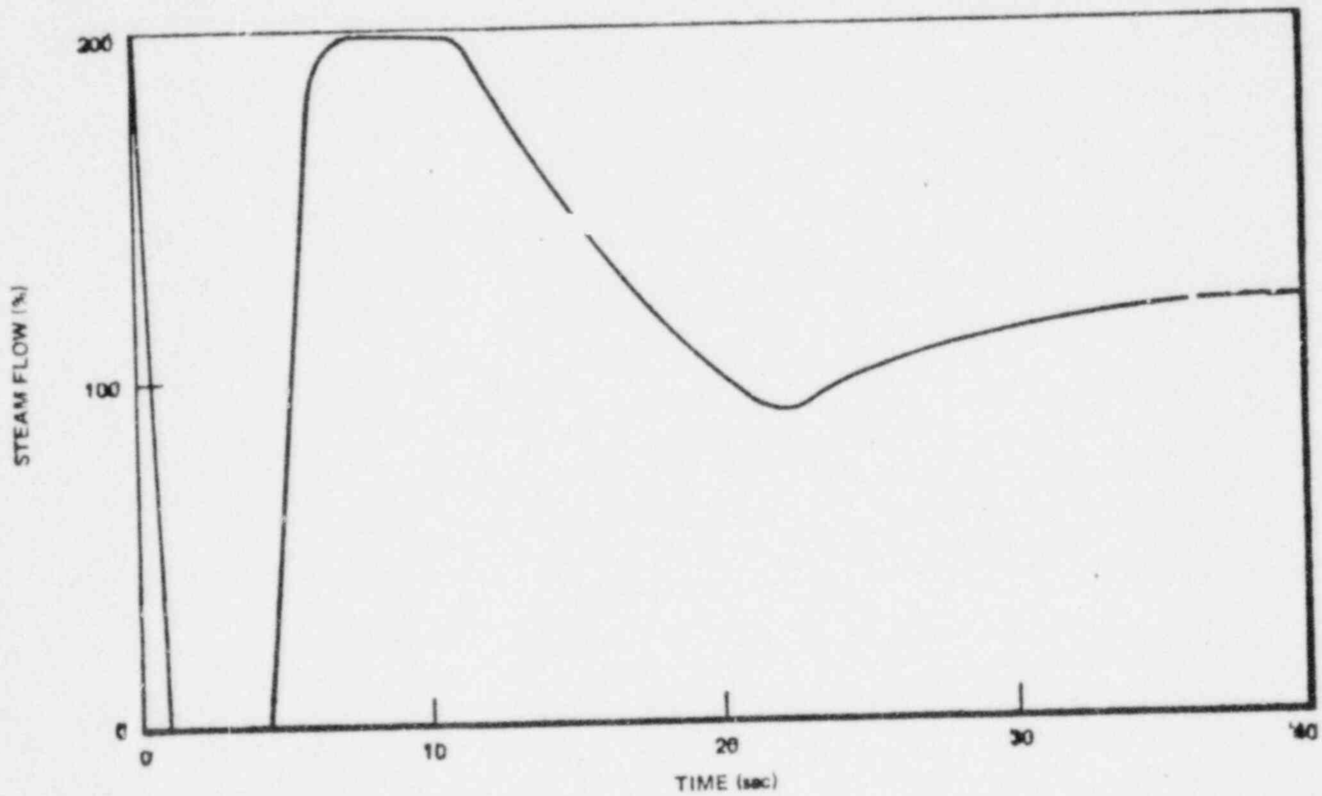

Figure 2. Vessel Steam Flow Transient Used in the Containment Response Analysis in an ATWS Load Rejection Without Bypass. Data Taken from the Safety Valve Sizing Analysis of Reference 1, Figure 15, Run 3.

attains a steady value at about 40 seconds. This is assumed to remain so until the liquid control becomes effective in the core and thereafter is assumed to linearly drop to the flow corresponding to decay heat. This is shown in Figure 3. The steam release into the containment sphere is the vessel flow minus the emergency condenser flow, as shown in Figure 3.

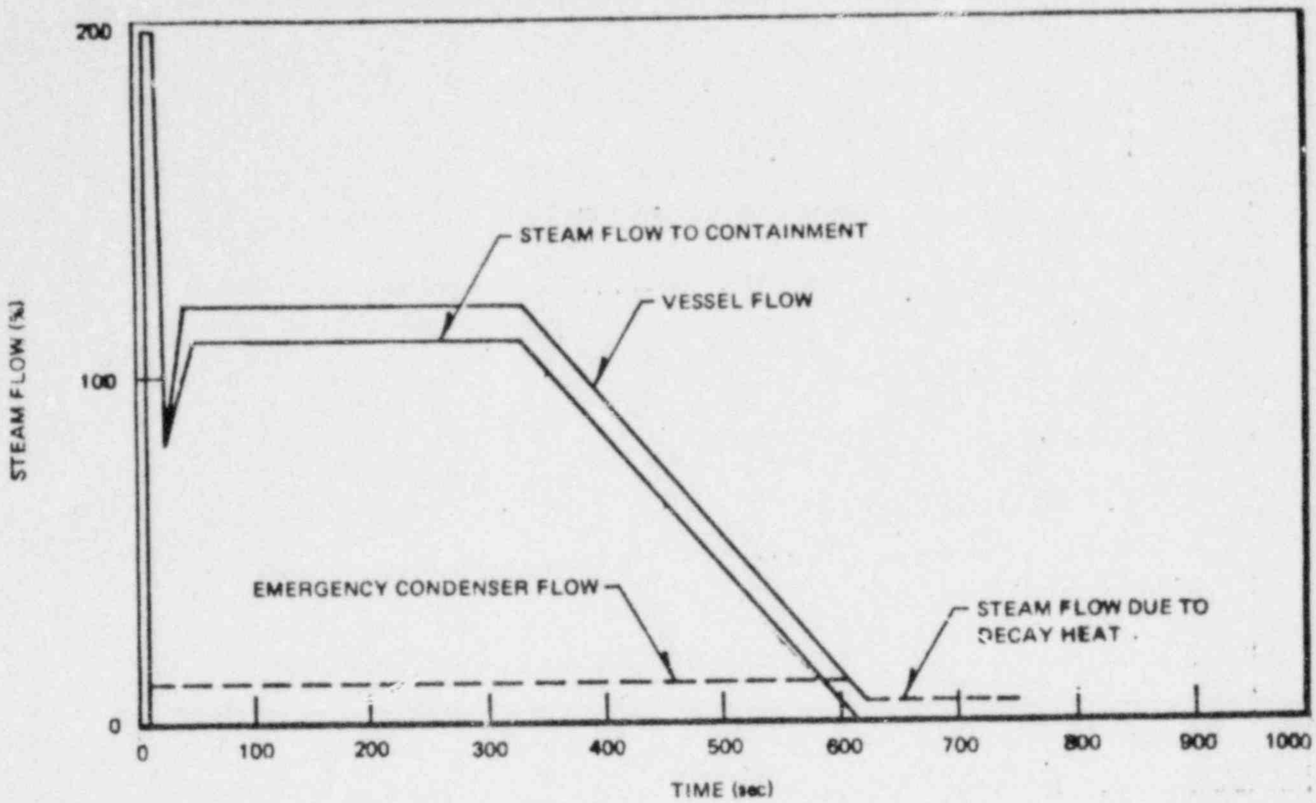

Figure 3. Vessel Steam Flow, Emergency Condenser Flow, and Steam Flow to Containment Used in the Containment Response Analysis in ATWS Load Rejection Without Bypass. (Initial Part of Vessel Flow Transient Replotted from Figure 2.)

The steam release into the containment sphere causes the containment temperature and pressure to rise. This rise is partially offset by the containment wall which absorbs heat from the air-water-vapor system inside. The containment wall temperature also rises by heat absorption. The containment sprays are assumed to be initiated by the operator at 5 minutes. After the steam release into the containment is terminated, the cooling effect of the spray recools the containment. The containment wall also recools later on losing heat as the containment sprays cool down the air-water-vapor system to a temperature below that of the wall.

The containment pressure and temperature responses are shown, respectively, in Figures 4 and 5. The containment wall temperature transient is also shown in Figure 5. The peak values of the containment pressure and temperatures are, respectively, 57.7 psig and 286°F.

The heat transfer coefficient between the containment air-water-vapor system and the wall is assumed to be a function of the air-to-water vapor weight ratio. The functional relationship that is used in the analysis is shown in Figure 6, which also shows the experimental data from Reference 5. The relationship used in the calculations is a simple modification of the data from Reference 5.

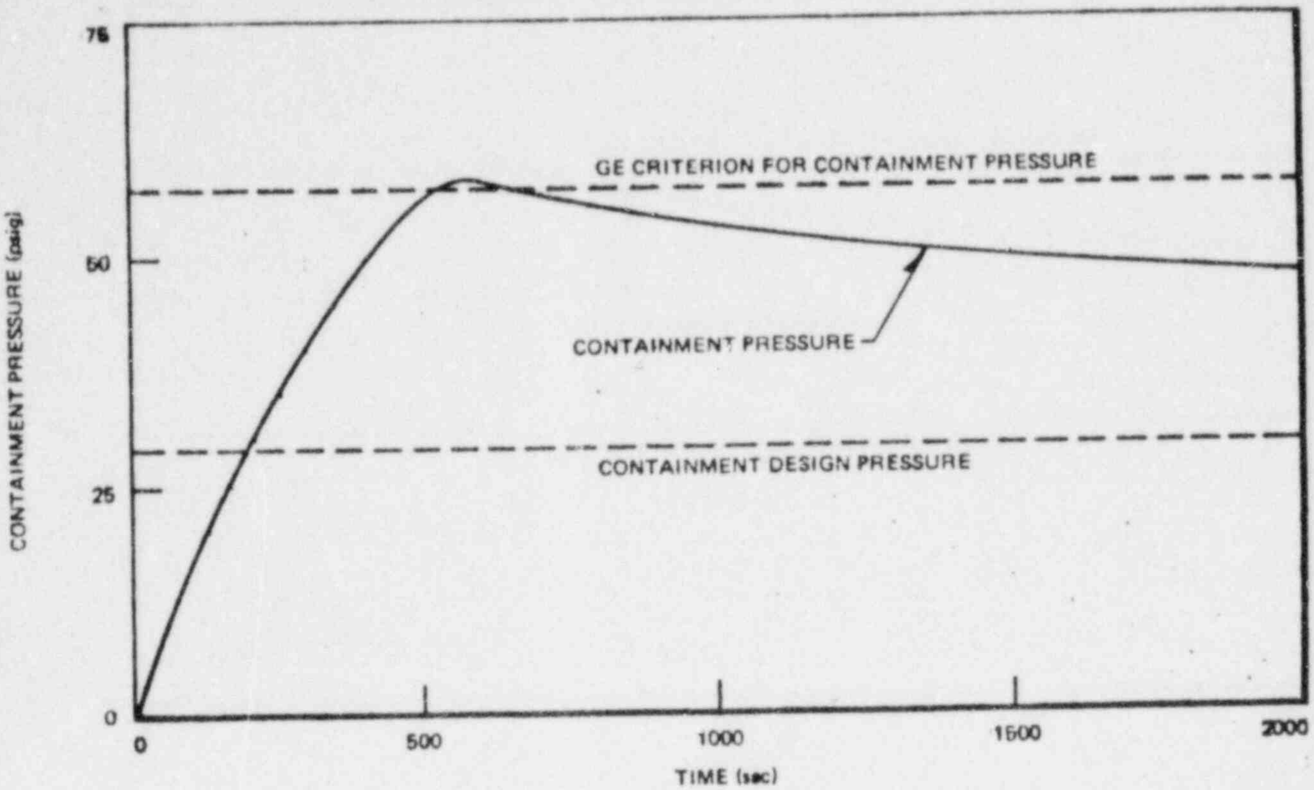

Figure 4. Containment Pressure Response in an ATWS Load Rejection Without Bypass

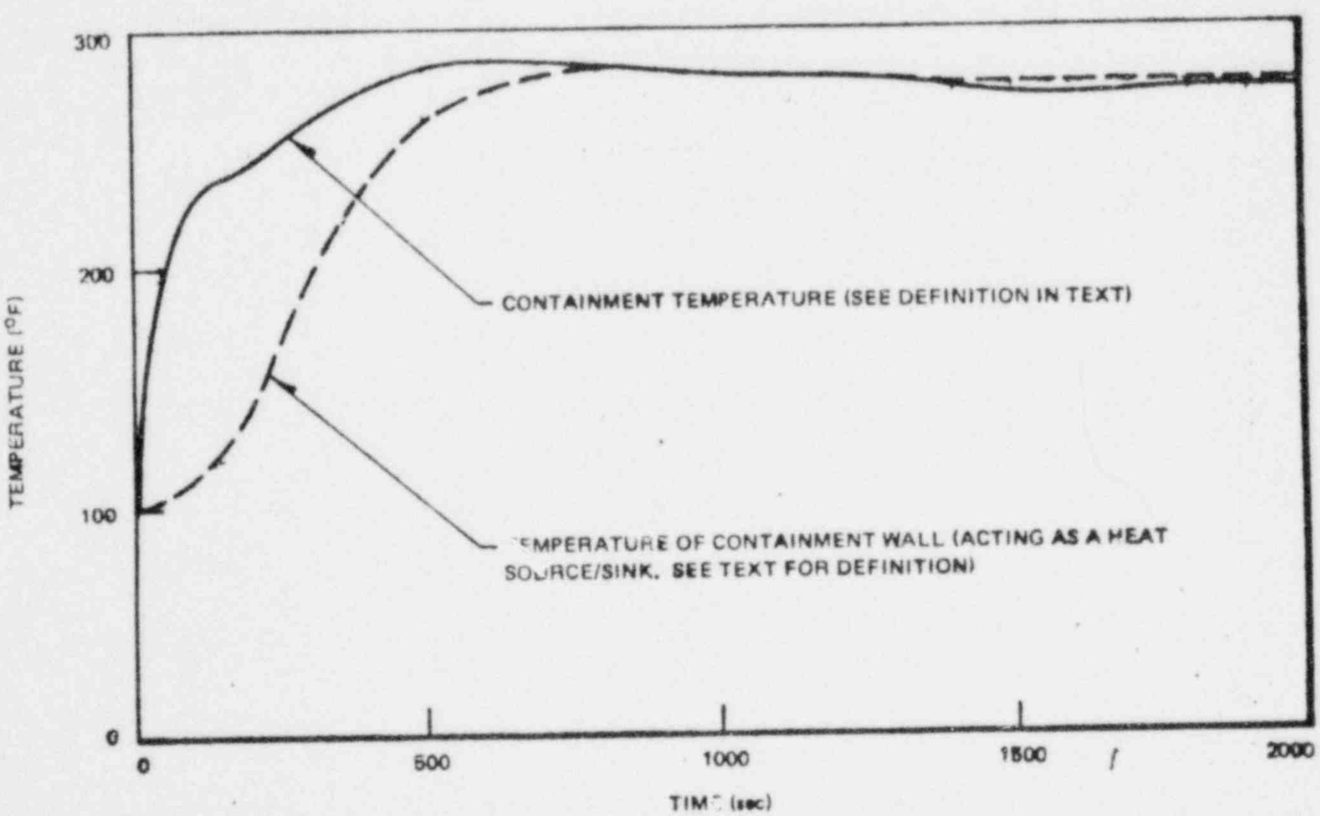

Figure 5. Containment Temperature and Containment Wall Temperature in an ATWS Load Rejection Without Bypass

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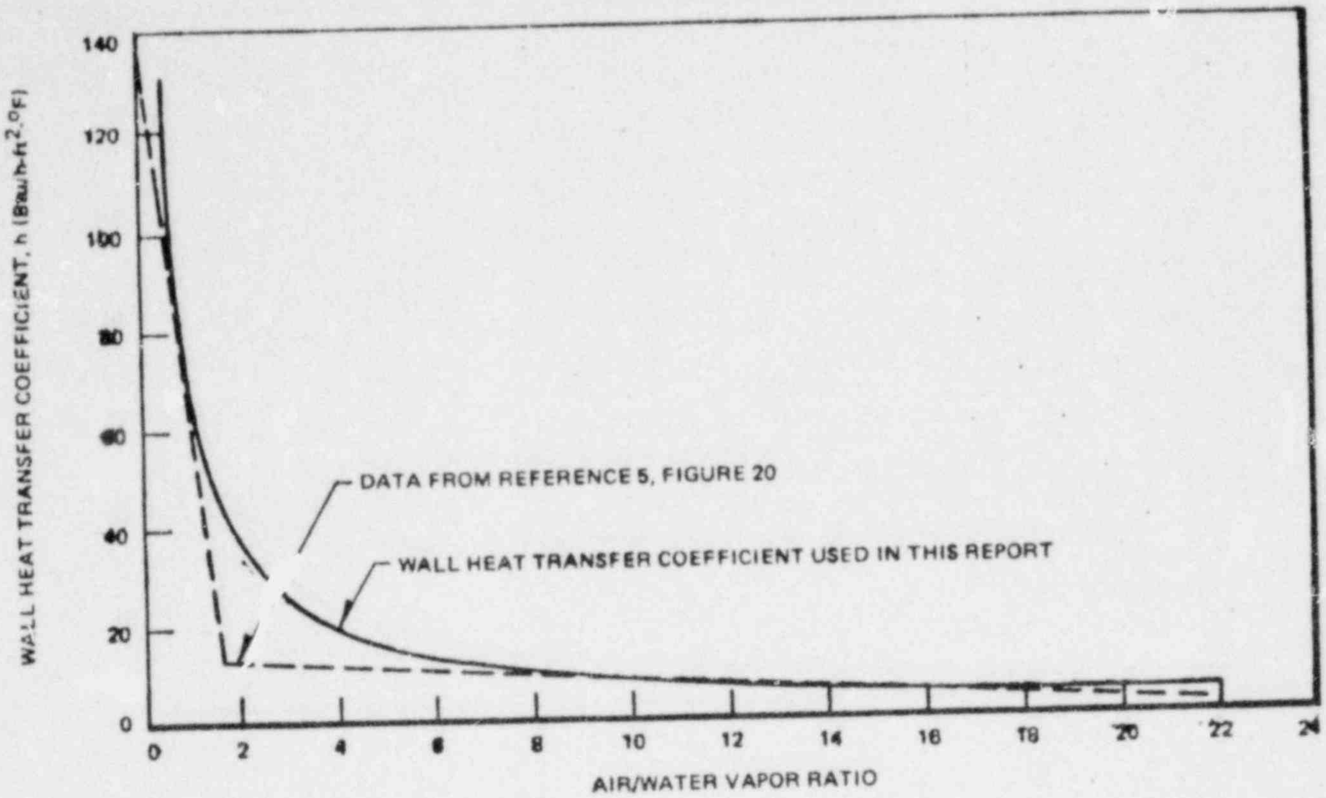

Figure 6. Containment Wall Surface Heat Transfer Coefficient as a Function of Air-to-Water-Vapor Ratio

6.4 COMPARISON TO WASH-1270

Appendix A, Paragraph II.C.1 of WASH-1270 requests comparison of three specified functions to specified analytical guides. This comparison is shown in Table 6-1.

Table 6-1
FULL LOAD REJECTION WITHOUT BYPASS AND WITHOUT SCRAM

Functional Comparison Parameter	WASH-1270 Comparison Value	General Electric Suggested Guide	Value By Analysis
Vessel Pressure (psig)	2040	2700	1587
Fuel Enthalpy (cal/gm)	280	280	<165
Cladding Oxidation (%)	17	17	<1
Containment Pressure (psig)	27	54	57.7

7. REFERENCES

1. R. J. Romeo and E. C. Eckert, *Transient Analysis Consumers Power Company Big Rock Point Plant*, General Electric Company, October 1962 (APED-4093).

2. *Consumers Power Company, Big Rock Point Plant, Technical Specifications*, (Revised to February 24, 1975).

3. *Final Hazards Summary Report, Big Rock Point Power Plant*, Consumers Power Company (November 1961).

4. Letter from James S. Rang of Consumers Power Company to Ray Fairfield of General Electric Company, dated September 11, 1975.

5. H. Uchida, A. Oyama, and Y. Togo, *Evaluation of Post-Incident Cooling Systems of Light Water Power Reactors*, Third International Conference on the Peaceful Uses of Nuclear Energy, pp. 93-103 (in particular, Figure 20), New York (1965).

APPENDIX A
PARAMETRIC AND SENSITIVITY STUDIES OF THE CONTAINMENT RESPONSE

The purpose of this parametric study is to make the analysis as inclusive as possible over the life of the plant. It also helps to cover certain uncertainties in the input data used for the evaluation of the containment response.

A.1 RECIRCULATION PUMP TRIP

Recirculation pump trip is an effective method of reducing the severity of an ATWS event that is being considered for many of the other BWR plants. Trip of the recirculation pumps reduces the core flow and, therefore, the core power (by increased core voids) and steam output into the containment. According to present plant operating procedures, the operator must immediately trip one recirculation pump on certain plant transients. Similarly trip of both the recirculation pumps can be initiated by the operator upon the occurrence of an ATWS. The containment peak pressure and temperature for recirculation pump trip initiated at various times after the ATWS event were calculated and are shown in Table A-1. The time evolution of the containment pressure and temperature responses are shown in Figures A-1 and A-2, respectively

Table A-1
CONTAINMENT RESPONSE TO ATWS: WORST REACTOR ISOLATION
(LOAD REJECTION WITHOUT BYPASS)

Initiation Time for RPT (sec)	Containment Peak Pressure (psig)	Containment Peak Temperature (°F)
6	25.0	234
60	27.5	240
120	31.0	247
150	33.0	251
180	35.1	254.5
300	43.9	269
No RPT	57.7	286

The vessel steam flow used for the containment response evaluation with recirculation pump trip was derived as follows.

The effect of tripping the recirculation pumps on the steady state operation of the reactor was studied in Reference 1 and shown in Figure 19 of that reference. There the effect of recirculation pump trip on the steam flow is to reduce it to 40% over about 12 seconds after a delay of about 2 seconds, and then to increase it to a steady-state value of about 50%. In the present case, the effect of the recirculation pump trip is assumed to decrease the vessel steam flow to 50% in 12 seconds after a delay of 2 seconds. The basis for this assumption is Figure 19 of Reference 1 and the fact that the containment response is not very sensitive to the details of the steam flow over short intervals of time (< 10 seconds). Thus the vessel steam flow with recirculation pump trip is given by

$$\begin{bmatrix}\text{Vessel Steam Flow}\\ \text{With Recirculation}\\ \text{Pump Trip}\end{bmatrix} = \begin{bmatrix}\text{Vessel Steam Flow}\\ \text{Without Recirculation}\\ \text{Pump Trip}\end{bmatrix} \times \begin{bmatrix}\text{(Multiplication Factor}\\ \text{Shown in Figure A-3)}\\ \end{bmatrix} \qquad \text{(A-1)}$$

A.2 STEAMING RATE

The long-term response of the nuclear boiler in ATWS depends, among other things, on the characteristics of the fuel which change with reloads. For containment considerations, this results in a change in the steam dump before reactor shutdown. To cover this aspect of ATWS transients, containment response was calculated with the vessel steam flow (shown in Figure 3) multiplied by a factor of 0.8 and 1.2. The resulting peak containment pressures and temperatures are shown in Table A-2.

Table A-2
CONTAINMENT RESPONSE TO ATWS: WORST REACTOR ISOLATION
(LOAD REJECTION WITHOUT BYPASS)

Containment Steam Flow Multiplication Factor	Containment Peak Pressure (psig)		Containment Peak Temperature (°F)	
0.8	43[a]	19.7[b]	267[a]	222[b]
1.0	57.7[a]	25.0[b]	286[a]	[illegible]34[b]
1.2	73.0[a]	30.5[b]	302[a]	246[b]

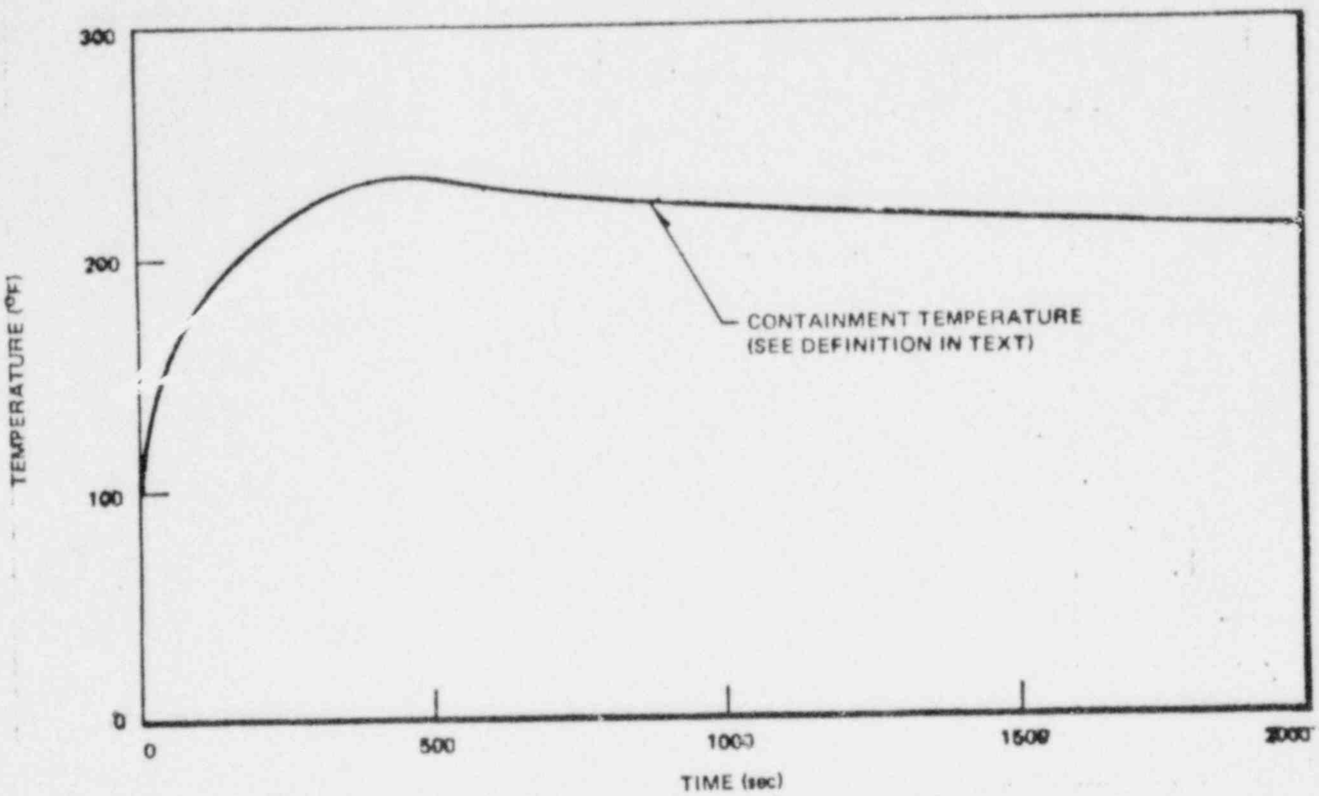

Figure A-1. *Containment Pressure Response in an ATWS Load Rejection Without Bypass, With Recirculation Pump Trip at 6 seconds*

POOR ORIGINAL

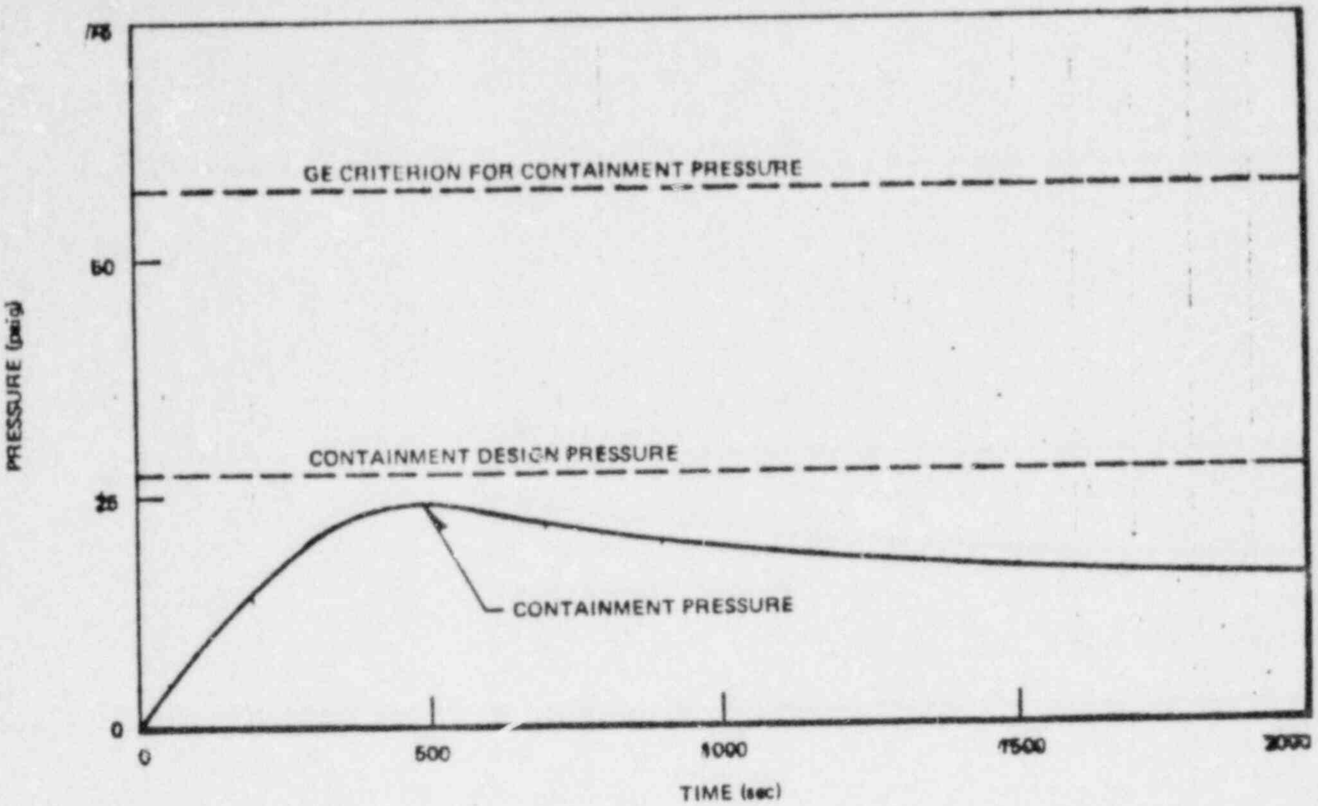

Figure A-2. Containment Temperature Transient in an ATWS Load Rejection Without Bypass, With Recirculation Pump Trip at 6 seconds

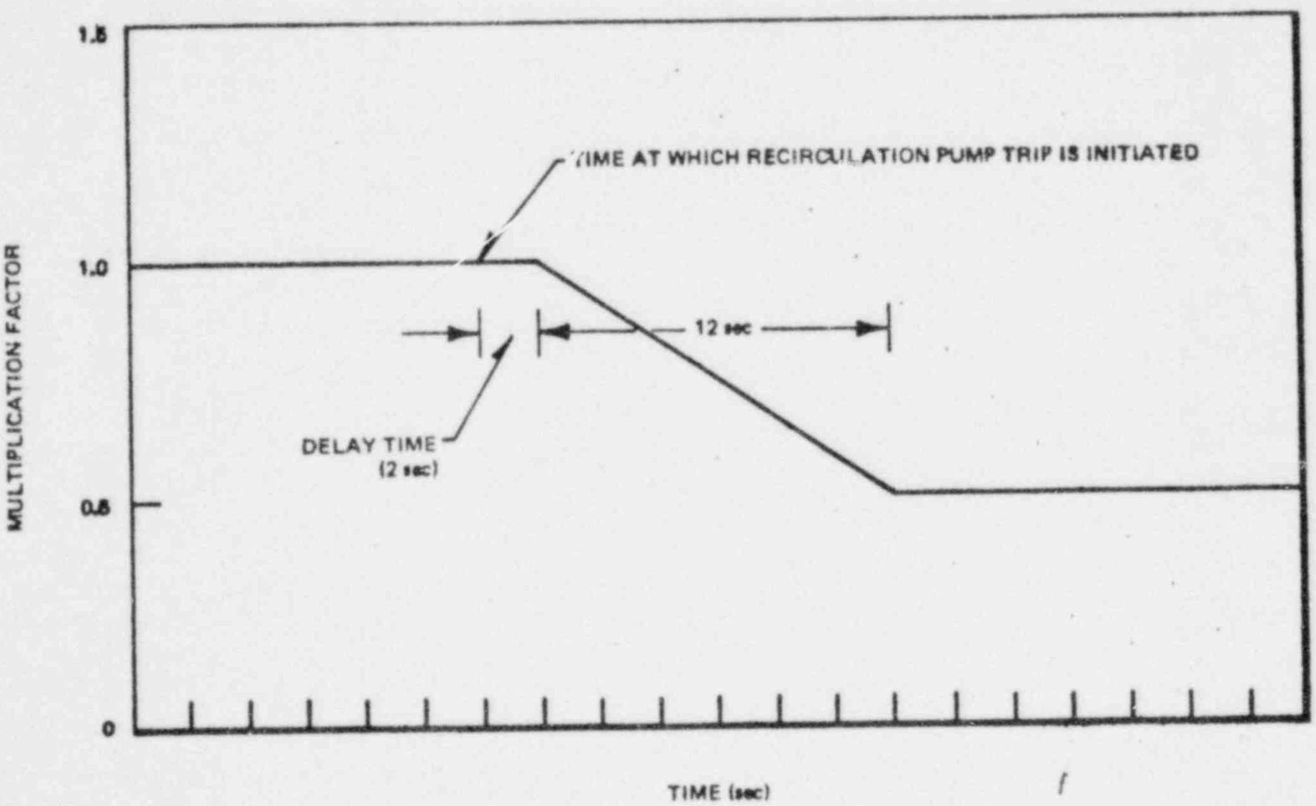

Figure A-3. Multiplication Factor Used in Equation A-1 for the Effect of Recirculation Pump Trip on Vessel Steam Flow. This Plot is Based on Figure 19 of Reference 1.

A.3 EFFECT OF EMERGENCY CONDENSER

The reduction in thermal load on the containment caused by the emergency condenser depends on the number of its tube bundles in operation. The effect of this on the containment peak conditions is shown in Table A-3.

Table A-3
CONTAINMENT RESPONSE TO ATWS: WORST REACTOR ISOLATION
(LOAD REJECTION WITHOUT BYPASS)

No. of Emergency Condenser Tube Bundles In Service	Containment Peak Pressure (psig)		Containment Peak Temperature (°F)	
2	57.7[a]	25.0[b]	286[a]	234[b]
1	61.7[a]	27.3[b]	291[a]	240[b]
None	66.0[a]	30.5[b]	295[a]	246[b]

[a] With no trip of recirculation pump
[b] With recirculation pump trip at 6 sec.

A.4 SLC INITIATION TIME

Historically, a delay time of 10 minutes has been assumed, following an unforeseen event, to take credit for operator action initiation or control of an emergency core cooling system. For the postulated ATWS event, however, the lack of scram would be evident within seconds. For this reason it would seem that initiation of the SLC system at less than 10 minutes could be justified. The base case results (shown in Figures 4 and 5) use a 5-minute initiation time. To show the effect of SLC initiation time, Table A-4 shows the peak containment pressures and temperatures for three different values of the initiation time.

Table A-4
CONTAINMENT RESPONSE IN ATWS: WORST REACTOR ISOLATION
(LOAD REJECTION WITHOUT BYPASS)

SLC Initiation Time (sec)	Containment Pressure (psig)		Containment Peak Temperature (°F)	
600	96.2[a]	37.6[b]	322[a]	259[b]
300	57.7[a]	25.0[b]	286[a]	234[b]
180	42.3[a]	20.5[b]	266[a]	233[b]

[a] With no trip of recirculation pumps
[b] With recirculation pump trip at 5 seconds

A.5 SENSITIVITY OF CONTAINMENT RESPONSE TO WALL HEAT TRANSFER COEFFICIENT

The heat transfer coefficient between the air-water-vapor mixture in the containment and the containment wall used in the base case results of Figures 4 and 5 was based on the experimental data from Reference 5. However, the experimental data of Reference 5 were for a vertical surface of 14-cm width and 30-cm height. The geometry of the containment sphere is thus different from that in Reference 5. For this and other reasons, the heat transfer characteristics in the two cases may be different. To provide a feeling for the effect of this on the containment response, analyses were made using half and twice the value of the heat transfer coefficient shown in Figure 6. The results are shown in Table A-5.

Table A-5
CONTAINMENT RESPONSE IN AN ATWS: WORST REACTOR ISOLATION (LOAD REJECTION WITHOUT BYPASS)

	Containment Peak Pressure (psig)		Containment Peak Temperature (°F)	
Containment Wall Heat Transfer Coefficient from Figure 6	57.7[a]	25[b]	286[a]	234[b]
Containment Wall Heat Transfer Coefficient Equal to Half the Value from Figure 6	60.1[a]	27.1[b]	289[a]	239[b]
Containment Wall Heat Transfer Coefficient Equal to Twice the Value from Figure 6	57.0	22.5	285	229

[a] With no trip of recirculation pumps.
[b] Recirculation pump trip at 6 sec.