DUANE ARNOLD ENERGY CENTER
SUPPRESSION POOL TEMPERATURE RESPONSE

July 1979

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1.0. INTROOUCTION

The Duane Arnold Energy Center (DAEC) takes advantage of the large thermal capacitance of the suppression pool during plant transients requiring safety/relief valve (SRV) actuation. The reactor pressure vessel (RPV) steam is piped through the main steam lines down the SRV discharge piping into the suppression pool, where it condenses, resulting in a temperature increase of the pool water, and an increase in the containment pressure. Most transients that result in relief valve actuations are of very short duration and have a small effect on the suppression pool temperature.

However, certain single and multiple failure events can be postulated which have the potential to discharge steam into the suppression pool for an extended period of time, significantly increasing the pool temperature. This may result in a situation where the suppression pool temperature and the ramshead discharge mass flux are such that the condensation stability limit may be approached. Stable condensation is expected for ramsheads if the suppression pool bulk temperature does not exceed 150°F when the ramshead mass flux (G) is greater than 40 lbm/sec-ft^2. The condensation phenomena is determined by the local temperature in the vicinity of the discharge device, whereas the calculations assume a bulk temperature. The bases for the 10°F temperature difference between bulk and local temperature and the condensation phenomena are given in Reference 2.

The Nuclear Regulatory Commission requested (Reference 1) that four postulated events, which have the potential to discharge steam into the suppression pool for an extended period of time, be analyzed to demonstrate that the condensation stability limit will not be exceeded at the Duane Arnold Energy Center. These four events are:

1. Stuck-open safety-relief valve (SRV) during power operation.

2. Stuck-open SRV during isolated hot standby.

3. Controlled RPV depressurization from isolated hot standby.

4. Automatic Depressurization System (ADS) activation following a small line break.

This document presents the results of the analysis of these events for the Duane Arnold Energy Center.

2.0 SUMMARY

The events mentioned in Section 1.0 were analyzed using licensing basis safety analysis values, and assume that the DAEC Technical Specifications are not violated, except as indicated in Assumption 16 in Section 3.3. Section 3.0 describes the calculational model, initial conditions, and assumptions that were used in the analyses. Section 4.0 presents the event descriptions and the complete analysis results. Section 5.0 presents the conclusions of the analysis.

The analyses results are summarized in Table 1. For each of the events analyzed, values of the maximum suppression pool temperatures when the ramshead discharge mass flux (G) is greater than the critical value of 40 lbm/ft^2-sec are given. Values of the discharge mass flux when the suppression pool temperature reaches the critical value of 150°F are also given. The table also shows the number of RHR loop(s) that were assumed operable and the number of SRV's that were opened to depressurize the reactor. The analysis results show that the condensation stability limit will not be exceeded for these events at the Duane Arnold Energy Center.

3.0 MODEL DESCRIPTION, INITIAL CONDITIONS AND ASSUMPTIONS

3.1 Model Description

3.1.1 Non-LOCA Events

To solve the transient response of the reactor vessel and suppression pool temperature due to the postulated events, a coupled reactor vessel and suppression pool thermodynamics model was used. The model is based on the principles of conservation of mass and energy and accounts for any possible flow to and from the reactor vessel and the suppression pool as shown in Figure 1.

The model incorporates a control volume, which includes the reactor pressure vessel (RPV) and the suppression pool. The RPV model is capable of tracking the reactor vessel water level and having a rate of change of temperature or pressure imposed on it. The various modes of operation of the residual heat removal (RHR) system have been simulated as well as the relief valves, HPCI, RCIC and feedwater functions. The model also simulates system setpoints (automatic and manual), operator actions and accepts as input the specific plant geometry and equipment capability.

3.1.2 Small Break Accident Model

In the Small Break Accident analysis, the mass and energy conservation laws are applied to a control volume which includes all of the reactor vessel contents and its walls. This control volume is subjected to the boundary conditions of decay heat input. The break and the safety-relief valve flow rates and the associated fluid enthalpies are derived from the state of fluid in the control volume undergoing the transient and the specified flow areas and location.

The time dependent break and safety-relief valve mass and energy flows are then input to another control volume containing the suppression pool. The pool temperature transient is obtained using the energy and mass balance equations on the suppression pool.

3.2 Initial Conditions

The following initial conditions were assumed in all of the analyses:

1. Operation at licensing bases safety analysis steam flow conditions. (DAEC = 105% NBR steam flow.)

2. Maximum condensate storage tank water temperature. (DAEC = 95°F.)

3. Maximum RHR heat exchanger service water temperature. (DAEC = 95°F.)

4. Suppression pool temperature at normal power operation Technical Specification limit (T_{op}). (DAEC = 95°F.)

5. Minimum Tech Spec suppression pool water volume. (DAEC = 58,900 cu. ft.)

6. Drywell air pressure at maximum of normal operating band. (DAEC = 1.4 psig.)

7. Drywell air temperature at normal drywell average. (DAEC = 135°F.)

8. Wetwell air pressure at maximum of normal operating band. (DAEC = -0.1 psig.)

3.3 Assumptions

The following assumptions were used in all of the analyses, except as noted in the individual event descriptions in Section 4.0 and in the Appendix.

1. Normal auxiliary power is available.

2. Normal automatic operation of the plant auxiliary systems (HPCI, RCIC, Core Spray, LPCI and ADS).

3. Control Rod Drive (CRD) flow maintained constant.

4. Duty of RHR heat exchangers based on maximum observed equilibrium crud buildup.

5. SRV capacities at 122.5% of ASME rated.

6. Licensed decay heat curve (May-Witt) for containment analysis (adjusted to account for delay between scram and isolation).

9. Motor-driven feedwater pumps were assumed to continue operating during the events.

10. Linear reduction of main steam flow rate from its initial flow rate at the start of reactor isolation to zero flow in 3.5 seconds (MSIV closure time).

11. In calculating the overall heat transfer coefficient of the vessel wall and internal structures, it is assumed that the heat transfer is dominated by conduction.

12. The heat transfer area of the reactor internals is obtained by assuming that they have the same metal thickness as that of the vessel wall, which is assumed to be 0.333 ft. uniformly.

13. The control volume of the reactor includes the reactor vessel, the recirculation lines, and the steam lines from the vessel to the inboard main steam isolation valves (MSIV), and the effects of feedwater flow.

14. The initial water level in the reactor vessel is calculated based on the assumption that the voids in the two-phase region collapse. Therefore, the ECCS volumes are based on the total liquid volume of the reactor vessel, the feedwater lines and the recirculation lines combined.

15. The specific heat of the reactor vessel and the internals is assumed to be 0.123 btu/lbm/°F. The metal density is assumed to be 490 lbm/ft^3.

16. The Duane Arnold Energy Center Technical Specifications are not violated during the events, except that:

    a) depressurization to avoid condensation instablity is not limited to the normal cooldown rate (100°F/hr) specified in Technical Specification 3.7.A.1.c.(4),

    b) during reactor isolation conditions, reactor depressurization is begun when the pool temperature reaches 110°F, rather than the temperature of 120°F given in Technical Specification 3.7.A.1.c.(4), and

c) since the SRV capacities are assumed to be 122.5% of ASME rated, the reactor pressure corresponding to the ramshead critical SRV discharge mass flux of 40 lbm/sec-ft^2 is 144.5 psig, rather than the value of 200 psig given in Technical Specification 3.7.A.1.c.(4).

These assumptions are justified in Section 5.2.

17. Operator actions are based on normal operator action times during the given event.

18. A stuck-open relief valve can be detected and the corresponding ramshead within the torus identified.

19. Additional safety/relief valves are manually opened, as necessary, to depressurize the reactor.

4.0 ANALYSIS

A detailed description of the event sequences and analysis results for the four events which have been analyzed is presented below. A chronological summary of the event sequences is presented in the Appendix.

4.1 Safety Relief Valve Oischarge During Non-LOCA Events

The first classification of SRV discharge events considered are those which occur under non-LOCA (i.e., non-accident), conditions. These events consider a stuck-open relief valve (SORV) condition and the manual operation of additional SRV's to depressurize the vessel.

4.1.1 Event 1: Stuck-Open Relief Valve From Power

4.1.1.1 Event Description

For this event, the reactor is initially operating at the power level corresponding to 105% NBR steam flow (1658 Mwt) and the suppression pool temperature is at 95°F. An SRV is postulated to stick fully open at this time (time zero of the transient analyzed). The plant operator is alerted that an SRV has opened by a drop in power output, as well as other plant parameter indications. The plant operator attempts to close the SRV but is unsuccessful. Therefore, the operator initiates a reactor scram at the time when the pool temperature has reached the maximum value permitted during power operation (110°F). As a result of the scram, the voids in the reactor collapse and the reactor water level drops momentarily below Level 2 of the reactor protection system, causing the main steam isolation valves (MSIV) to close. At 10.5 seconds after scram the MSIV's are assumed to be fully closed. During this transient the motor-driven feedwater pumps are assumed to operate continuously.

The pool temperature rises due to the steam discharge from the SORV into the suppression pool. Cooling of the suppression pool, utilizing the RHR heat exchangers in the pool cooling mode, is initiated three (3) minutes after the pool temperature reaches the maximum value permitted during normal power operation (95°F). At the time of isolation the plant operator is assumed to open additional SRV's to depressurize the reactor vessel such that, should the

bulk pool temperature reach 150°F, the ramshead mass flux at the discharge point is below 40 lbm/sec ft^2. A ramshead discharge mass flux of 40 lbm/sec ft^2 corresponds to a reactor vessel pressure of 144.5 psig (159.2 psia).

4.1.1.2 Analysis Results

The analysis was performed assuming:

1) Both RHR loops are available - Event 1(a).

2) Only one RHR loop is available - Event 1(b).

The results are shown in Figures 2 and 3, and are summarized in Table 1.

The vessel depressurization and the resulting suppression pool temperature increase with both RHR loops available is shown in Figure 2. Time zero in the figure corresponds to the time of reactor pressure vessel isolation. The stuck open relief valve initially opened approximately 200 seconds prior to the time of TCV closure.

At the time of isolation (suppression pool temperature approximately 110.8°F and approximately 3 minutes after SORV) additional SRVs are manually opened by the operator. The vessel pressure drops quickly as steam is discharged through the SRVs and incoming feedwater cools the vessel liquid. At 57 seconds the vessel water level approaches the high water level trip setpoint of the feedwater control system. The rate of feedwater injection then begins to decrease, and consequently the rate of vessel depressurization decreases. At about 333 seconds

the vessel pressure drops below 144.5 psig, which corresponds to the critical SRV discharge mass flux of 40 lbm/sec-ft^2.

The manually opened SRVs are closed by the operator after the vessel pressure drops below 144.5 psig and before it reaches 100 psig. This is to reduce the vessel cooldown rate (°F/hr). The vessel will continue to depressurize due to the one stuck-open relief valve. However, it is possible that the one SORV may close at this lower vessel pressure if the operator attempts to close the valve.

The analysis results for the SORV at full power event with only one RHR loop available (Figure 3) are identical to those of the case with two RHR loops available up to the time that the RHR loops are turned on in the pool cooling mode. This occurs at 3 minutes after the time that the stuck open relief valve initially opened. With only one RHR heat exchanger available, the pool temperature increases slightly more rapidly than when both RHR loops are available. However, as shown in Figure 3, the opening of four additional SRVs at the time of isolation is sufficient to depressurize the reactor to below 144.5 psig before the pool reaches 150°F.

Additional pool temperature margin could be obtained by manually opening more than four SRVs to depressurize the reactor vessel. However, the analysis results show that opening only four SRVs is sufficient to avoid the condensation instability threshold, and also minimizes the containment loadings and the thermal duty on the vessel due to the resulting depressurization rate.

Additional analysis showed that if the operator waited until 10 minutes after the occurrence of the SORV to open additional SRV's, the condensation instability threshold could not be avoided. Therefore, it is necessary that the operator manually open additional SRV's when the pool temperature reaches 110°F (approximately 3 minutes after SORV).

It should be noted that for Event 1 the manual actuation of the Automatic Depressurization System (ADS) may not be sufficient to avoid the condensation instability threshold. This is due to the fact that only four valves are assigned the ADS function. If the SORV is an ADS valve, activation of ADS will open only three additional valves rather than the required four valves. Therefore, the operator must manually open four additional SRV's when the pool temperature reaches 110°F in order to avoid the condensation instability threshold during this event.

4.1.2 Event 2: Stuck-Open Relief Valve From Hot Standby

4.1.2.1 Event Description

In Event 2 the initial pool temperature is 95°F and the reactor power level is initially at 105% NBR steam flow (1658 Mwt) at the time of reactor scram.

A transient occurs which results in reactor scram and isolation. The reactor is held in an extended hot standby condition during which the reactor pressure is maintained at 920 psig by operating the SRV's intermittently. The steam flow from these SRV actuations causes the suppression pool temperature to rise. Cooling of the suppression pool, utilizing the RHR heat exchangers in the pool cooling mode, is initiated three (3) minutes after the pool temperature reaches the maximum value permitted during normal power operation (95°F). When the pool temperature is at 110°F, the plant operator begins the required reactor depressurization. At this point in the transient, a relief valve is postulated to stick open. Additional SRVs are therefore opened and maintained open to depressurize the reactor vessel so that the ramshead mass flux is below the critical value of 40 lbm/sec-ft^2 when the bulk pool temperature reaches 150°F.

4.1.2.2 Analysis Results

For Event 2, the analysis was performed assuming:

1) Both RHR loops are available - Event 2(a).

2) Only one RHR loop is available - Event 2(b).

The results are shown in Figures 4 and 5, and are summarized in Table 1.

The vessel pressure transient for Events 2(a) and 2(b) are very similar, the major difference being the time that it takes for the pool temperature to increase to 110°F. When the pool temperature reaches 110°F, the operator begins the depressurization of

the vessel. The suppression pool temperature responses are also similar, varying primarily by the difference in pool cooling capability assumed during the events.

With both RHR loops available, the SORV from Hot Standby Event is as follows. At time zero the reactor is scrammed and isolated. Following the scram and isolation, the vessel pressure (Figure 4) increases to the lowest relief valve set-point and is maintained at this value by the automatic opening of the SRVs. At about 300 seconds the plant operator begins manually operating the SRVs to maintain the vessel pressure at 920 psig.

As shown in Figure 4, the pool temperature increases rapidly to about 106°F during the period of automatic relief valve operation. When the plant operator begins manually operating an SRV (at 300 seconds) to depressurize the vessel to 920 psig, the pool temperature rapidly increases to 110°F during the 55 seconds that the valve is open.

When the suppression pool temperature has reached 110°F (at about 355 seconds after isolation), the plant operator begins a controlled depressurization of the reactor pressure vessel (RPV). A safety-relief valve is assumed to stick open at this time. The plant operator immediately opens, and maintains open, three additional SRV's. Therefore, starting at 355 seconds the RPV experiences a rapid depressurization due to the steam flow through the four open SRVs (i.e., the SORV plus three (3) additional valves).

At approximately 782 seconds the vessel pressure drops below, and subsequently remains below, the critical vessel pressure of 144.5 psig. The pool temperature at the time the vessel pressure drops below the critical value (144.5 psig) is 149°F. The manually opened SRVs are closed by the operator after the vessel pressure drops below 144.5 psig and before it reaches 100 psig.

The assumption that only one RHR loop is available during an SORV from Hot Standby (Event 2(b)) does not conform to the plant licensing basis. Also, the operator would not go into hot shutdown with only one RHR loop available. The following analysis of this event is presented for information only.

The discussion of Event 2(a) also applies to Event 2(b), with the following exceptions. As shown in Figure 5, if only one RHR loop is available various events occur slightly earlier in time. The vessel depressurization, which begins when the pool temperature reaches 110°F, is initiated at 354 seconds after isolation. The critical vessel pressure of 144.5 psig is reached at 782 seconds. The suppression pool temperature at the time the vessel pressure drops below 144.5 psig is 149°F.

Additional pool temperature margin could be obtained by manually opening more than three SRVs to depressurize the reactor vessel. However, the analysis results show that opening only three SRVs are sufficient to avoid the condensation instability threshold during this event.

Additional analysis showed that if the operator waited until the pool temperature reached 120°F to begin the reactor depressurization, the condensation instability threshold could not be avoided. Therefore, it is necessary that the operator manually open the additional SRV's when the pool temperature reaches 110°F.

4.1.3 Event 3: Controlled RPV Depressurization from Hot Standby

The event sequence for Event 3 is identical to that given for Event 2, except that the operator performs the vessel depressurization at a controlled cooldown rate (°F/hr). However, because the controlled cooldown rates would be very high, an operator controlled cooldown rate would be difficult to maintain over the short period of time of this transient. Therefore, the depressurization procedure required for Event 3 is the same as that required for Event 2. The operator is required to depressurize the RPV by opening three additional SRV's when the pool temperature reaches 110°F.

4.2 Safety Relief Valve Discharge During LOCA Event

4.2.1 Event 4: Small Line Break with ADS

4.2.1.1 Event Description

This event assumes that a small break occurs in the recirculation system inlet line. The size of this postulated liquid break is chosen to result in the highest suppression pool temperature when the ramshead discharge mass flux is $\geq$ 40 lbm/sec-ft^2.

In this event, the reactor is assumed to be automatically scrammed on a high drywell pressure signal (2 psig), which occurs at approximately 6 seconds after the small line break. Isolation is subsequently initiated automatically on low water Level 2. The actuated single active failure is the failure of the High Pressure Coolant Injection (HPCI) system. Therefore, the vessel pressure after the small liquid break remains nearly constant at the lower setpoint of the relief valves (1110 psia). Since no safety grade high pressure make-up system is available (credit is not taken for RCIC since it is not safety grade), the vessel water level continues to drop as mass is lost through the break and through the SRVs. Also, credit is not taken for the feedwater system, since it is not safety grade. At 120 seconds after the vessel water level reaches Level 1 of the Reactor Protection System (RPS), the ADS is automatically activated and rapidly depressurizes the vessel to well below the critical pressure required to produce a discharge mass flux at the ramshead of 40 lbm/sec-ft^2.

4.2.1.2 Analysis Results

To define the limiting break size, both a small line break area of 0.05 ft^2 and an intermediate line break with a break area of 0.1 ft^2 were considered. The results of these analyses showed that the 0.05 ft^2 break yields a slightly higher pool temperature (145°F) than does the 0.1 ft^2 break (142°F) when the critical discharge mass flux of 40 lbm/sec-ft^2 is reached. Therefore, the small line break was found to be the limiting case. The results of the analysis for the 0.05 ft^2 break area are presented in Figure 6.

The SBA with ADS transient sequence is as follows. At time zero the reactor is assumed to be scrammed on a high drywell pressure signal, which occurs at approximately 6 seconds after the line breaks. Following scram, the reactor isolates automatically on low water Level 2. Following isolation, the vessel pressure peaks briefly and then decreases and oscillates about the lowest relief valve set-point until the time when the Automatic Depressurization System (ADS) actuates at approximately 240 seconds after the time of scram. The suppression pool temperature, which is conservatively taken to be 97°F at the time of scram, rises gradually during this period as both the break flow and relief valve flow enter the pool. Upon actuation of the three available ADS valves, the vessel experiences a rapid depressurization and the suppression pool experiences a corresponding temperature increase. At approximately 440 seconds after scram, the low pressure Emergency Core Cooling Systems (ECCS) automatically actuate and slightly increase the rate of vessel depressurization as the cool ECCS water is pumped into the vessel. The pool temperature reaches 145°F at the time when the vessel pressure drops below 144.5 psig.

5.0 CONCLUSIONS

5.1 Conclusions

The results of the analysis are summarized in Table 1. These results support the following conclusions:

1. The condensation instability threshold is avoided in Event 1 (SORV during full power operation) by opening four additional SRVs when the pool temperature reaches 110°F. This occurs approximately three (3) minutes after the SORV initially sticks open.

2. The condensation instability threshold is avoided in Event 2 (SORV during isolated hot standby) by opening three additional SRVs when the pool temperature reaches 110°F.

3. The condensation instability threshold is avoided in Event 3 (Controlled Depressurization from Isolated Hot Standby) by opening three additional SRVs when the pool temperature reaches 110°F.

4. The condensation instability threshold is avoided in Event 4 (Small Break Accident with ADS) without any operator action. The ADS system will automatically actuate during this event and depressurize the RPV such that the condensation instability threshold is avoided.

5.2 Justification for Exceptions to the Technical Specifications

The analysis of the events described in this document was performed using licensing basis safety analysis values, and conformed to the current DAEC Technical Specifications, except that: 1) the required cooldown rate exceeded the "normal cooldown rate" given in Technical Specification 3.7.A.1.c.(4); 2) the temperature at which RPV depressurization is begun was assumed to be 110°F, rather than the value of 120°F given in Technical Specification 3.7.A.1.c(4); and 3) the reactor pressure corresponding to the critical SRV discharge mass flux of 40 lbm/sec-ft^2 was assumed to be 144.5 psig, rather than the value of 200 psig given in Technical Specification 3.7.A.1.c.(4). The justification for the use of these exceptions to the Technical Specification is as follows:

For the events analyzed, a cooldown rate in excess of the normal rate (100°F/hr) is required to avoid the condensation instability threshold. The required cooldown rates for both the SORV from power (Event 1) and the SORV from isolated hot standby (Event 2) are also in excess of the highest cooldown rate used in the design of the reactor pressure vessel. The required cooldown rates will cause a slight amount of fatigue damage to the reactor pressure vessel each time they occur. The additional fatigue damage resulting from the postulated events is approximately equal to that due to a normal startup and shutdown cycle. However, if T-quenchers are installed at the end of the current operating cycle, the cumulative fatigue damage resulting from the occurrence of any of these postulated events during the remainder of the current cycle will be acceptably small.

The analysis of Events (1) and (2) showed that it was necessary to begin RPV depressurization when the pool temperature reaches 110°F in order to avoid the condensation instability threshold with ramshead devices installed at the end of the SRV discharge lines. Therefore, the analyses in this document assumed that RPV depressurization was begun when the pool temperature reached 110°F, rather than the value of 120°F given in Technical Specification 3.7.A.1.c.(4).

The justification for the third exception to the Technical Specifications, the use of an RPV critical pressure of 144.5 psig, is based on SRV test data which indicates that actual SRV capacities may approach 122.5% of ASME rated.

The pressure of 144.5 psig is the reactor vessel pressure corresponding to the critical SRV discharge mass flux of 40 lbm/sec-ft^2 when the SRV capacities are assumed to be 122.5% of ASME rated (Figure 7). This assumption was made in the present analysis because it conservatively represents the actual SRV discharge mass flow rate into the pool. The ASME rated capacity of a valve is less than the actual valve capacity. This is conservative for pressure vessel

analyses, but is non-conservative for the present analysis, in which the pool temperature rise is dependent on the actual steam flow into the pool. This justifies the use of SRV capacities at 122.5% of ASME rated in the present analysis.

5.3 Justification For Exceptions to Reference (1) Analysis Assumptions

Reference (1) requested that figures be provided which depict the reactor pressure, safety/relief valve (SRV) discharge mass flux, and suppression pool bulk temperature versus time for the following events which are based on current Technical Specification limits:

(a) Stuck-open SRV during power operation assuming reactor scram at ten minutes after the suppression pool reaches a bulk pool temperature of 110 F and all RHR systems are operable.

(b) Same events as in (a) above with only one RHR train operable.

(c) Stuck-open SRV during hot standby assuming an initial 120°F bulk pool temperature and only one RHR train operable.

(d) Automatic Depressurization System (ADS) activated following a small line break assuming an initial 120°F bulk pool temperature and only one RHR train operable.

(e) Primary system is isolated and depressurized at a rate of 100°F per hour with an initial 120°F bulk pool temperature and only one RHR train operable.

The following table shows the correspondence between the Reference (1) events (above) and the present analysis:

Reference (1) Event	Event No. (this document)
(a)	1(a)
(b)	1(b)
(c)	2(a),2(b)
(d)	4
(e)	3

The analyses presented in this document conformed to the Reference (1) event scenarios, except where they were in violation of the plant Technical Specifications, or where they would result in the violation of the condensation instability threshold. These exceptions are discussed below.

Event (a)

The analysis in this document assumed that the reactor is scrammed when the suppression pool reaches a bulk pool temperature of 110°F, rather than 10 minutes after the bulk pool temperature reaches 110°F. Reactor scram at 110°F is based on Technical Specification 3.7.A.1.c.(3).

> The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit of 95°F.

The present analysis assumes that this Technical Specification is not violated. The pool temperature alarm notifies the operator that the pool temperature is approaching 95°F. The pool temperature indicator provides the operator with the water temperature around the pool.

Event (b)

Same comments as for Event (a) above.

Event (c)

The analysis in this document assumes that the SRV sticks open during hot standby when the bulk pool temperature reaches 110°F. If it is assumed that the SRV sticks open when the bulk pool temperature reaches 120°F (as specified in Reference 1) then the condensation instability threshold could not be avoided for this event. Therefore, the analysis in this document showed that it was necessary to begin depressurization when the pool temperature reaches 110°F, at which time an SRV is assumed to stick open.

The current analysis of this event assumes that 1) an SRV fails in the stuck open position, and 2) the main condenser is unavailable as a heat sink. The assumption of an additional failure, such as one RHR loop, is highly improbable, and violates the plant licensing basis.

The assumption in Reference (1) that only one RHR loop is available does not conform to the plant licensing basis. Also, if only one RHR loop was available, the operator would not go into hot shutdown. An analysis of this event, assuming that only one RHR loop is available, was performed. However, the results are presented for information only, and should not be used for licensing purposes.

Event (d)

The present analysis assumes that the bulk suppression pool temperature is 95°F, rather than 120°F, when the Small Line Break Accident occurs. This is in accordance with Technical Specification 3.7.A.1.c.(1):

Maximum water temperature during normal power operation - 95°F.

The pool temperature alarm will alert the operator that the pool temperature is approaching 95°F. The operator is required to initiate pool cooling before the pool temperature reaches 95°F. In the present analysis, it is assumed that the RHR loops are on for pool cooling at 3 minutes after the pool temperature reaches 95°F.

The present analysis was performed with both RHR loops available. The assumption that only one RHR loop is available does not conform to the plant licensing basis, or to 10CFR50 Appendix K requirements.

Event (e)

Because the calculated cooldown rates were very high, an operator controlled cooldown rate would be difficult to maintain over the short period of time of this transient. Therefore, the procedure required for Event (e) is the same as that required for Event (c). The operator is required to depressurize by opening three additional SRV's when the pool temperature reaches 110°F.

6.0 REFERENCES

1. Letter, G. Lear (NRC) to Iowa Electric Light and Power Company, "Suppression Pool Temperature Transients," (Part A), dated December 9, 1977.

2. "Steam Vent Clearing Phenomena and Structural Response of the BWR Torus (Mark I Containment)," NEDO-10859, April 1973.

TABLE 1

SUMMARY OF RESULTS

DAEC POOL TEMPERATURE RESPONSE

Event Description	Event No.	No. of SRV'S Manually Opened	No. of RHR Loops	Discharge Mass* Flux (G) at Pool Temp = 150°F	Maximum Pool Temp (°F)* at G ≥40 lbm/sec-ft²
Stuck Open Relief Valve at Power	1 (a)	4	2	40	150
	1 (b)	4	1	40	150
Stuck Open Relief Valve From Isolated Hot Standby	2(a), 3(a)	3	2	38	149
	2(b), 3(b)	3	1**	39	149
Liquid SBA w/ADS	4	None	N/A	N/A	145

* Values Rounded to Nearest Whole Number

** This event does not conform to the plant licensing basis, and is presented for information only. Also, the operator would not go into Hot Shutdown with only one RHR loop available.

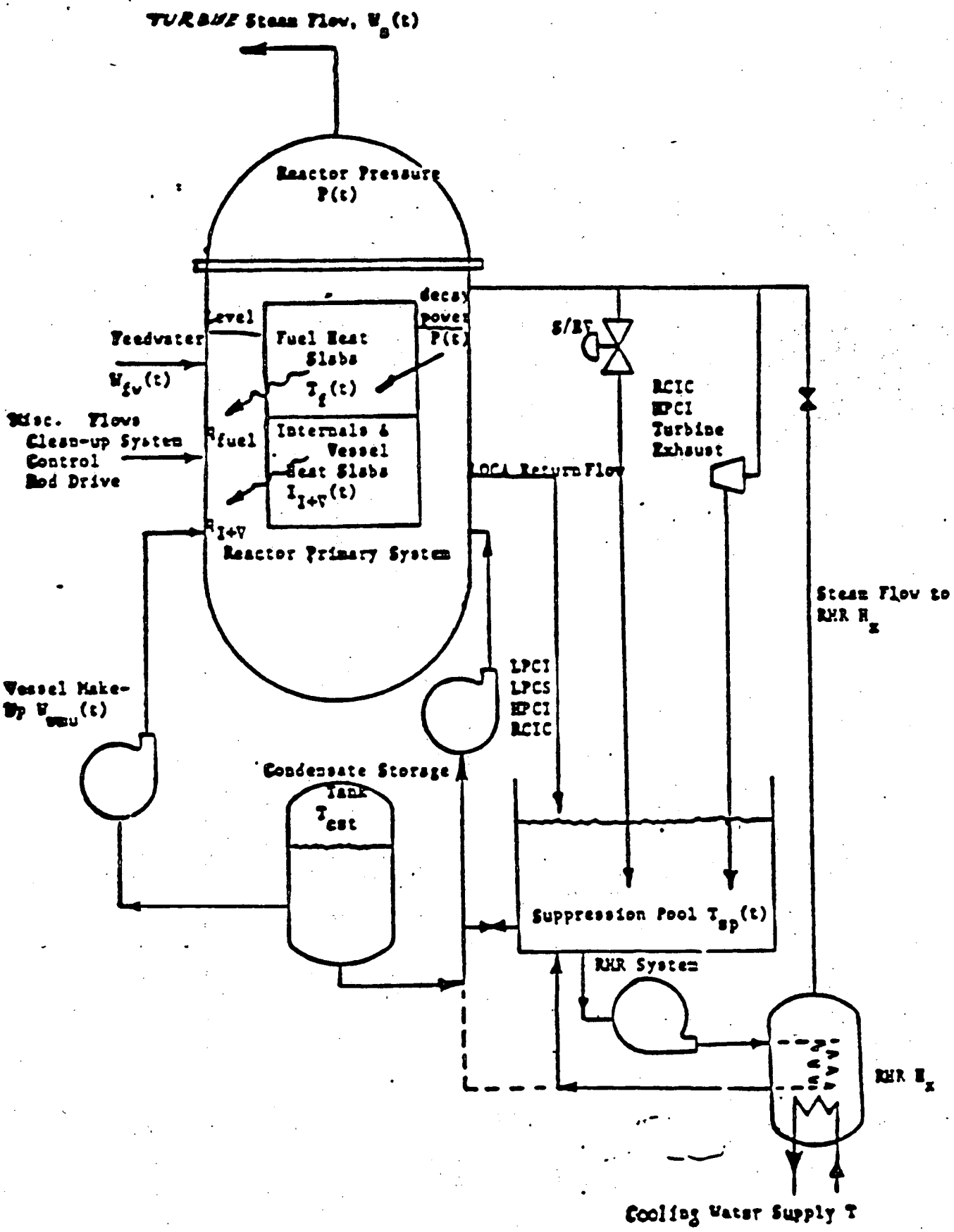

Figure 1. Model Schematic of Reactor and Containment System - Non-LOCA Events

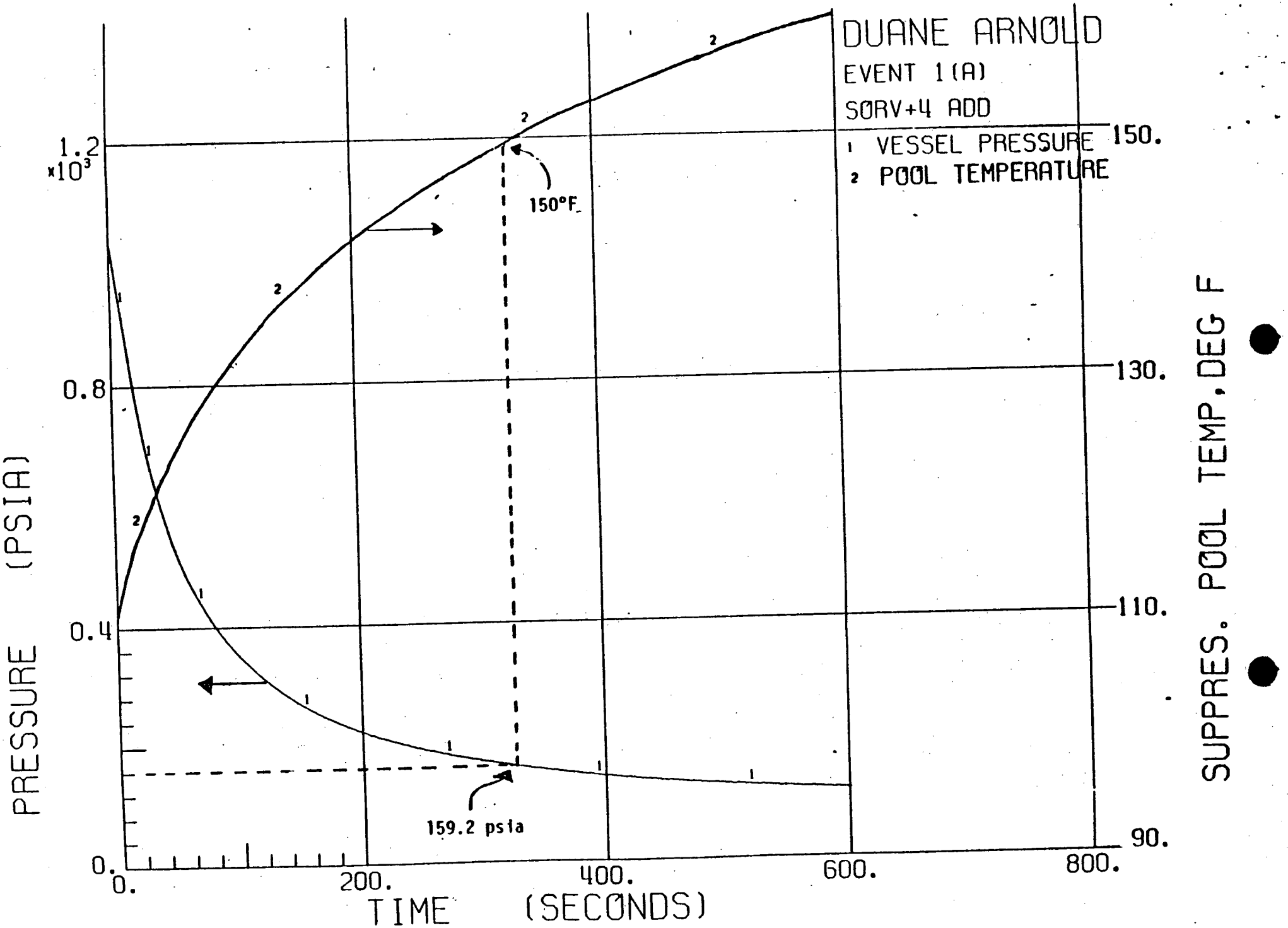

Figure 2. Stuck-Open Relief Valve at Power - 2 RHR Loops

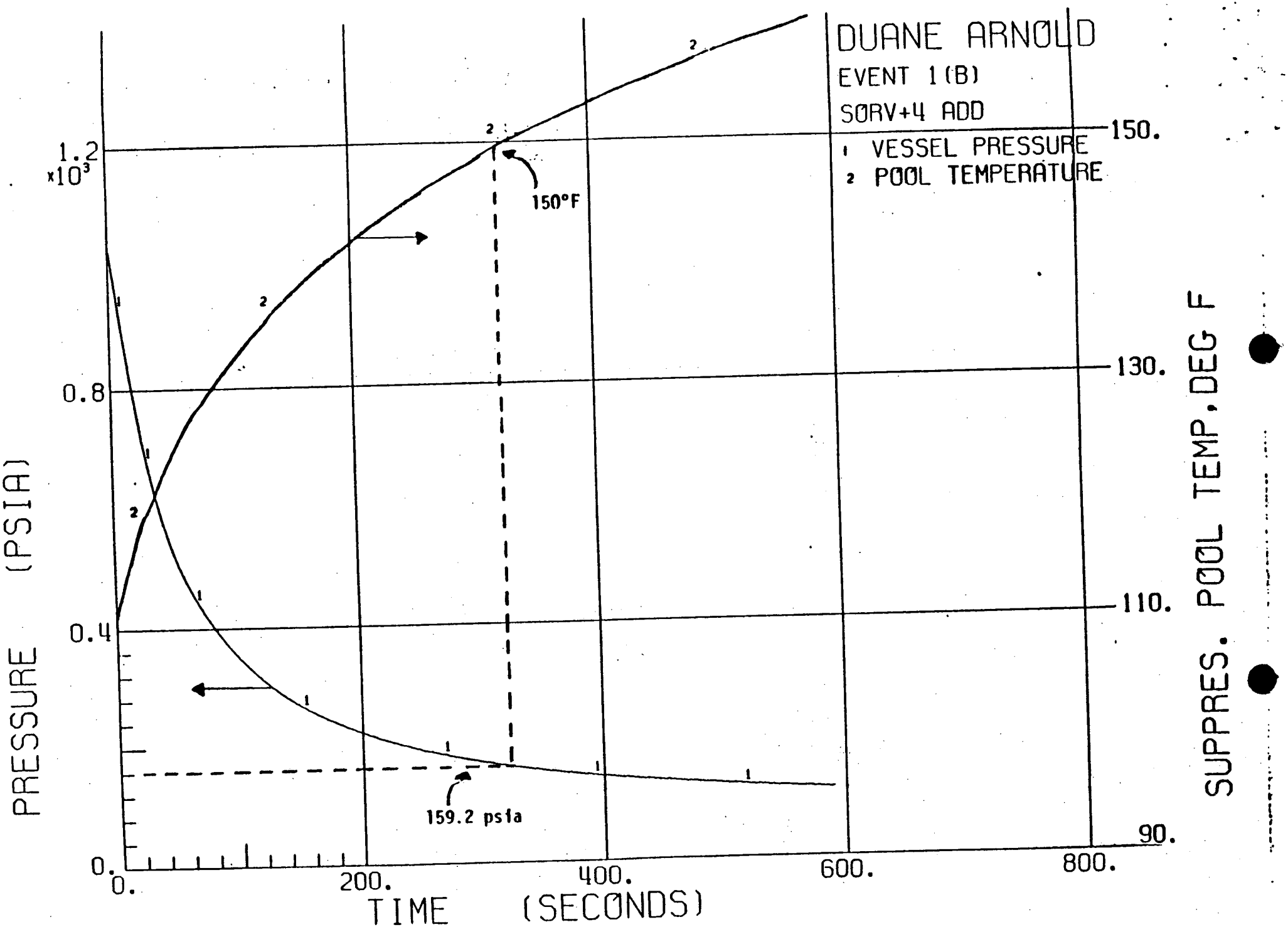

Figure 3. Stuck-Open Relief Valve at Power - 1 RHR Loop

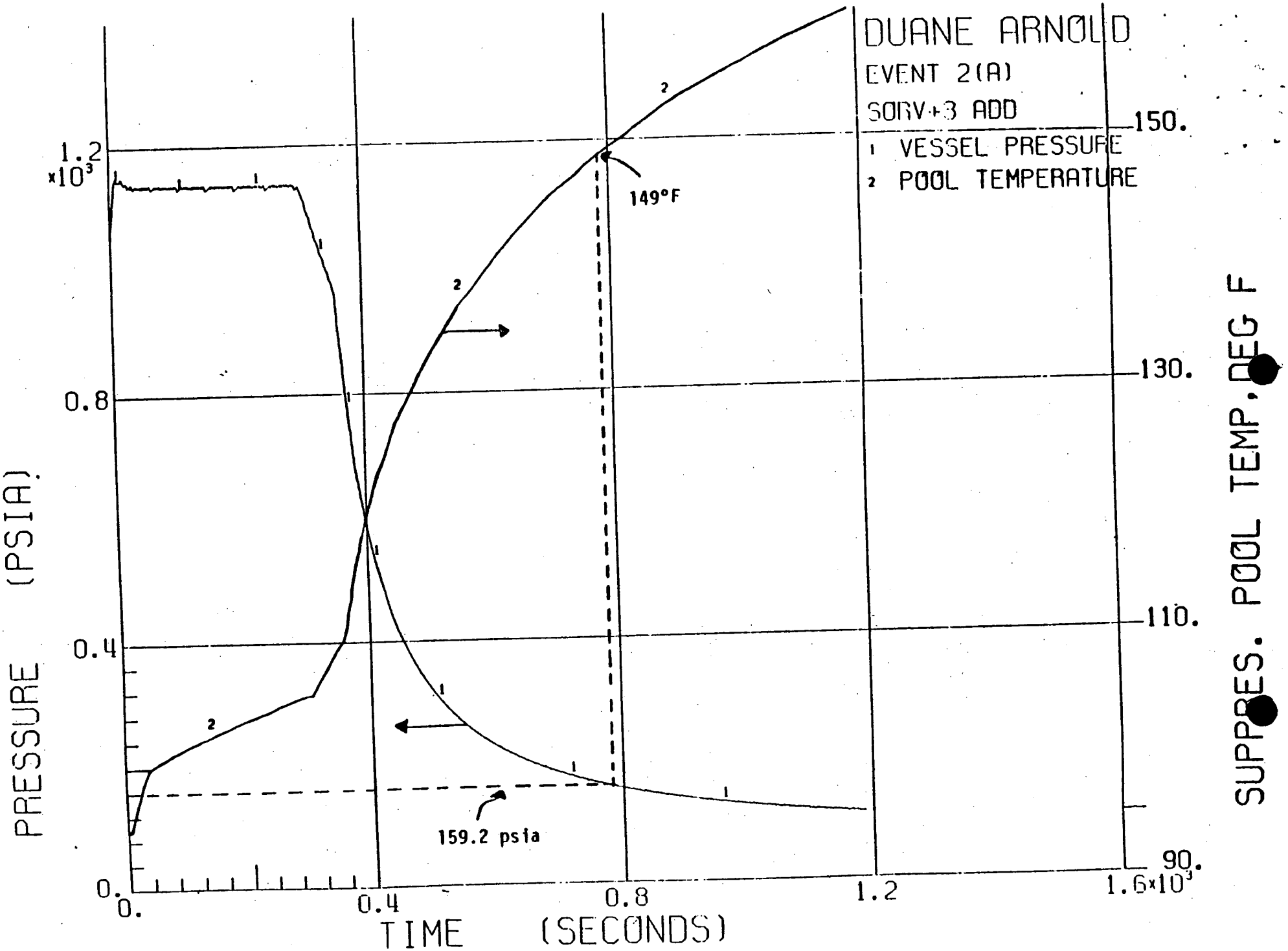

Figure 4. Stuck-Open Relief Valve From Isolated Hot Standby - 2 RHR Loops

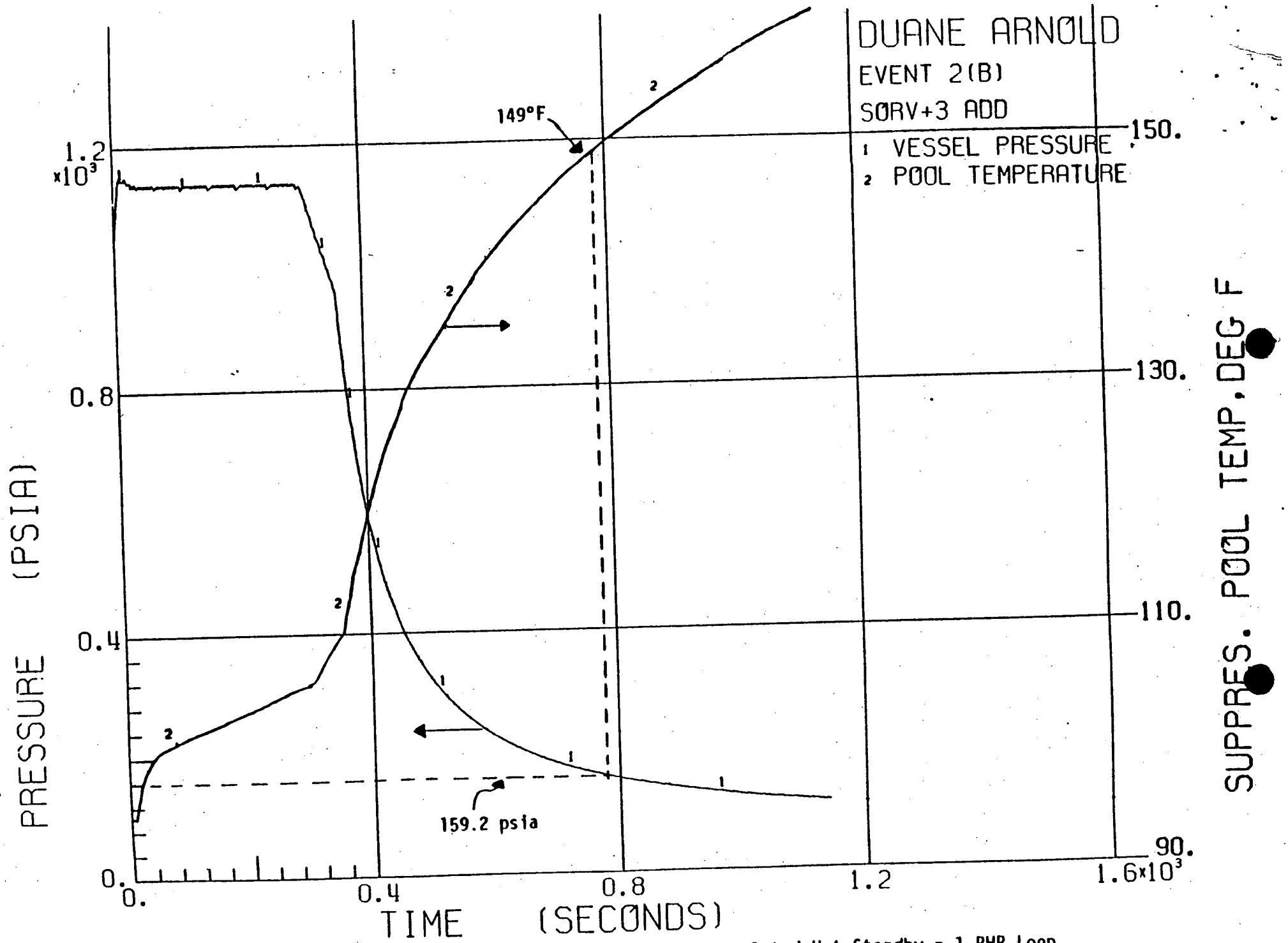

Figure 5. Stuck-Open Relief Valve From Isolated Hot Standby - 1 RHR Loop

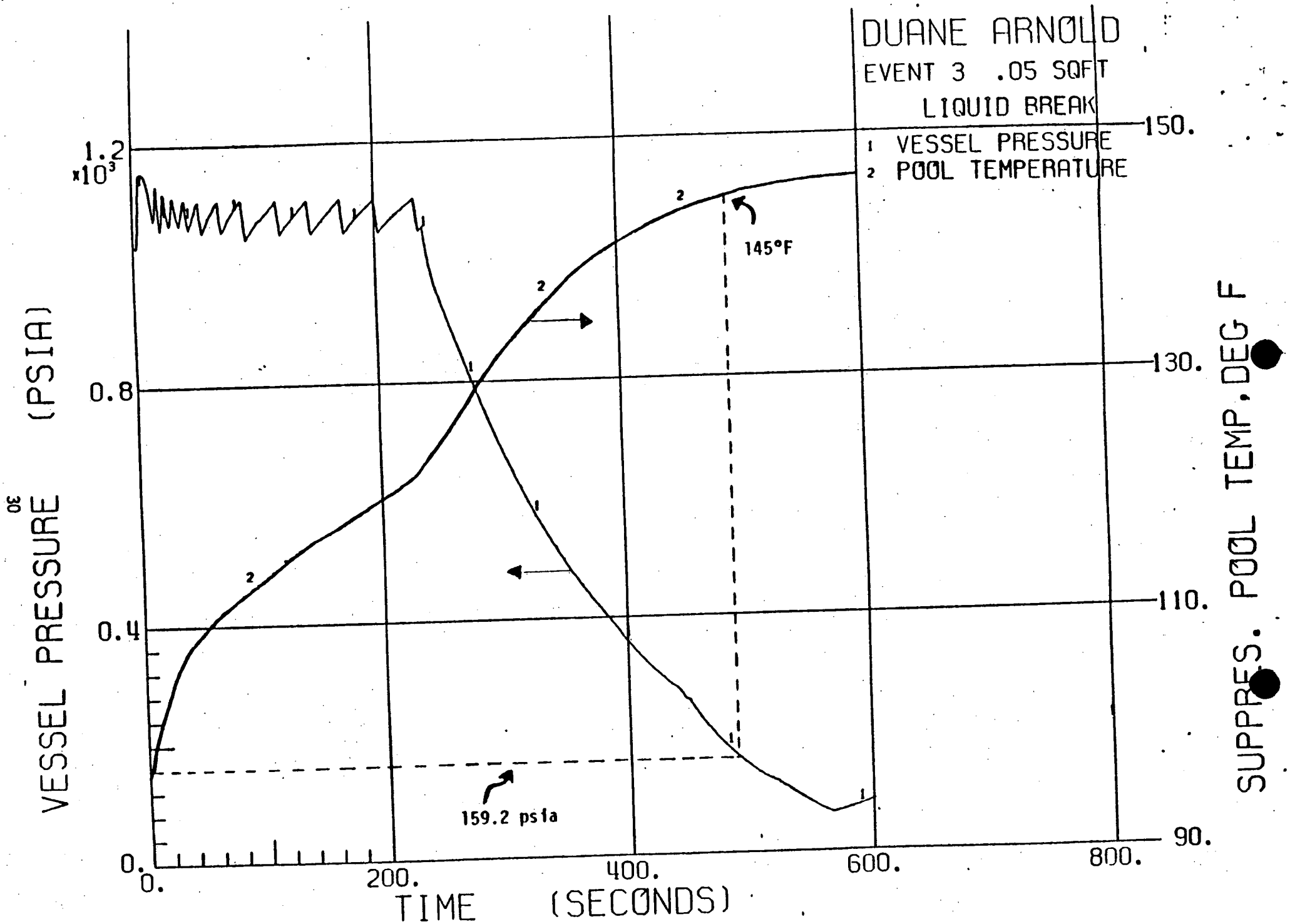

Figure 6. Liquid SBA With ADS

DUANE ARNOLD SUPPRESSION POOL TEMPERATURE RESPONSE

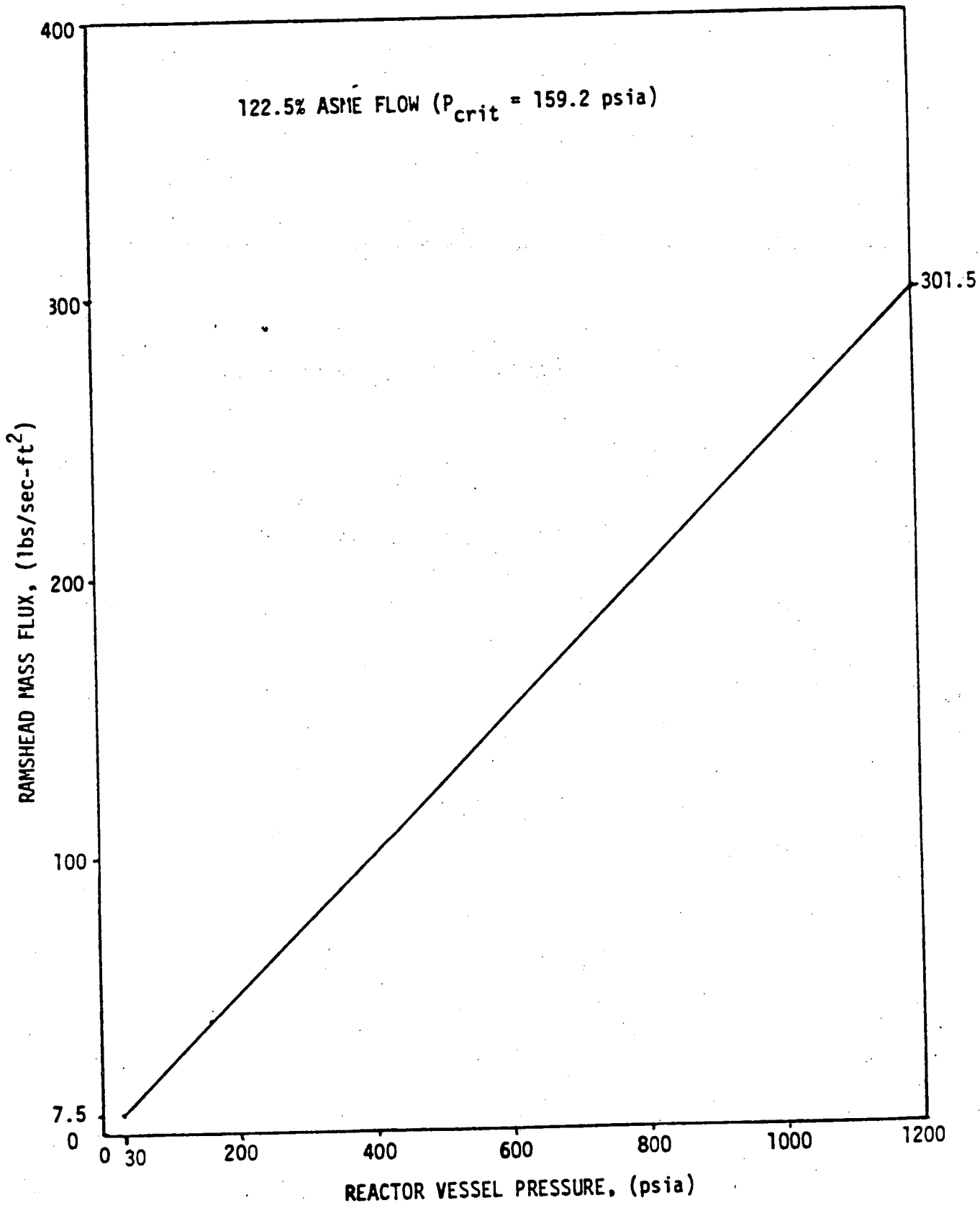

Figure 7. Ramshead Discharge Mass Flux vs. Reactor Vessel Pressure

APPENDIX

EVENT DESCRIPTIONS

EVENT 1* - STUCK-OPEN RELIEF VALVE FROM POWER OPERATION

Event Sequence

Time	Temp	Event Description
#t_a = 0.0	T_{op}	Pool temperature alarm at normal power operation, Tech Spec limit (95°F). Initiate actions to turn RHR loop(s)* on for pool cooling. SRV fails open.
t_a + 3 minutes**		RHR loop(s)* on for pool cooling and torus spray.
t_s	T_s	Reactor Scram*** (T_s = 110°F. for DAEC)
t_s + 10.5 seconds		Reactor isolated from main condenser (assuming automatic isolation on low water Level 2).
t_s + 10.5 seconds		Four additional SRV(s) manually actuated to depressurize the reactor.

Time t_s and the number of SRV's to be manually actuated by the operator to be determined by analysis.

---

* Corresponds to Event 1(a) if two RHR loops available and to 1(b) if one RHR loop available.

** The operator can complete the actions necessary to turn the RHR loop(s) on within three minutes.

*** Mode switch in Shutdown.

\# The bulk suppression pool temperature is assumed to be 95°F when an SRV fails open. This is the maximum pool temperature allowed during normal power operation (Technical Specifications Section 3.7.A.1). Also Section 3.7.A.1 specifies that the reactor shall be scrammed from any operating condition when the suppression pool temperature reaches 110°F.

EVENT 2* STUCK-OPEN RELIEF VALVE FROM ISOLATED HOT STANDBY

Event Sequence

Time (Min.)	Event Description
#$t_a = t_s = 0.0$	A transient has occurred, which causes the reactor to scram and isolate. The suppression pool temperature is at the normal limit (95°F). The operator initiates actions to turn RHR loop(s)* on for pool cooling.
t_a + 3 minutes**	RHR loop(s)* on for pool cooling and torus spray.
$t_a<t<t_o$	Reactor pressure maintained using SRV's.
t_o	Operator begins reactor pressure vessel depressurization by opening 3 additional SRV(s). Single SRV sticks open at 110°F.

---

* Corresponds to Event 2(a) if two RHR loops available and to 2(b) if one RHR loop available.

\# The bulk suppression pool temperature is assumed to be 95°F rather than 120°F when the reactor is scrammed. This is the maximum pool temperature allowed before pool cooling would begin.

** The operator can complete the actions necessary to turn the RHR loop(s) on within three minutes.

EVENT 3* ISOLATION AND REACTOR DEPRESSURIZATION

Event Sequence

Time (Min.)	Event Description
#$t_a = t_s = 0.0$	Reactor isolation and scram. Pool temperature at normal limit (95°F). Initiate actions to turn RHR loop(s)* on for pool cooling.
t_a + 3 minutes**	RHR loop(s)* on for pool cooling and torus spray.
$0<t<t_c$	Reactor pressure maintained using SRV's (intermittent operation).
t_c	Initiate cooldown using SRV's at 110°F.

---

* Corresponds to Event 3(a) if two RHR loops are available and 3(b) if one RHR loop is available.

** The operator can complete the actions necessary to turn the RHR loop(s) on within three minutes.

\# The bulk suppression pool temperature is assumed to be 95°F rather than 120°F. This is the maximum pool temperature allowed before pool cooling would begin.

EVENT 4* SMALL BREAK ACCIDENT WITH ADS

Event Sequence

Time (Min.)	Event Description
0.0	SBA occurs during normal plant operation**
t_s	Reactor scram on high drywell pressure.
t_i**#	Main steam line isolation initiated automatically on low water Level 2.
t_i + 3.5 seconds***	MSIVs fully closed.
t_g	Feedwater flow to reactor stops.
t_d	Water level drops to Level 1.
t_d + 120 seconds	ADS automatically activates, depressurizing the reactor vessel.

No operator actions assumed, event runs to completion.

The suppression pool temperature versus discharge mass flux is determined by the analysis.

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* This event requires that the HPCI System fails; otherwise ADS would not be activated. Additional assumptions for this accident event are:

1. Loss of normal auxiliary power.
2. Limiting small break which results in the highest pool temperature when the SRV discharge mass flux $\geq$ 40 lbm/ft^2 sec.

** The bulk suppression pool temperature is assumed to be 95°F rather than 120°F when the SBA occurs.

*** The MSIV closure time is 3.5 seconds.

**# Approximately 7 seconds is required for the water level to drop to Level 2 following reactor scram.