NUREG-0803 Environmental Qualification

Justification for Interim Operation

Susquehanna Steam Electric Station

Units 1 and 2

Docket No. 50-387

Pennsylvania Power and Light Company

September, 1982

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I. Introduction

A. Purpose

The purpose of this report is to demonstrate that interim operation of Unit 1 of the Susquehanna Steam Electric Station (SSES) is justified.

This study will show that one path of achieving and maintaining a cold shutdown condition is available using only Class 1E equipment which is environmentally qualified to the NUREG-0803 environment; or for which an acceptable justification for operation without being qualified has been identified. This study will also show that a means of SDV leak detection is available for which an acceptable justification for interim operation has been identified.

Each component which is required for the SDV break mitigation or detection (both on and off the single cold shutdown path), and not qualified for the break environment is evaluated to determine whether an appropriate justification for interim operation of the plant can be ascertained for that component.

B. Groundrules

A single path to cold shutdown must be determined for the SDV break identified in NUREG-0803. Random single failures of environmentally qualified equipment are not assumed. The only failures of equipment considered are those which may result from the SDV break.

The single Cold Shutdown and Break Detection (CSBD) path includes the minimum systems and equipment essential to emergency reactor shutdown, containment and reactor heat removal, and prevention of significant release of radioactive material to the environment; as well as, a SDV leak detection method that can provide a prompt and unambiguous notification of the leak.

All components required to mitigate or identify a SDV break, and required to function in the CSBD path must either be environmentally qualified to the break environment or "justified." A component is environmentally qualified if it meets the required Category of NUREG-0588 and the environment qualification requirements of NUREG-0803 Section 5.3 (Based on a Susquehanna SES specific environment). All equipment that is environmentally qualified to the SDV break conditions is not evaluated further in this study. The justification criteria and detailed component justification are discussed in part IV of this report.

C. Methodology

First, a study was performed to determine the environment (temperature, pressure, radiation, humidity) which would exist in the reactor building at SSES as a result of the SDV break postulated in NUREG-0803. This analysis was performed to supplement PP&L's response to NUREG-0803, and is referenced in that report (PLA-1132, dated June 18, 1982). Next, those safety actions (including break detection) necessary to achieve a cold shutdown condition from the SDV break were established; and the primary and auxiliary systems required to perform those safety functions were identified. This resulted in a list of system/components which would be used to mitigate or identify the break (this list is presented in Section III - A, and Appendix C). From this list, a preferred CSBD path was selected for further review, noting which components are subject to the break environment.

Those mitigation/identification (M/I) Components in the SDV environment which will not be environmentally qualified (Per NUREG's-0588 and 0803) were analyzed in detail with respect to their safety functions. All not qualified M/I Components were evaluated to determine whether interim operation of the plant is justified. This evaluation is presented in part IV. Qualified components are not evaluated further. Where a given component is not required for the CSBD path and is not qualified, interim operation is justified since the component is not on the CSBD path; however, the impact of failure of such a component upon the ability to achieve and maintain cold shutdown was considered. For those components which are on the CSBD path and are not qualified, an acceptable justification for using that component must exist (See Section IV-B).

The CSBD path is considered successful if it is demonstrated that all required M/I Components are either qualified or justified, and that failure of unqualified, unjustified components not in the preferred CSBD path will not affect the plant's ability to achieve and maintain a safe cold shutdown condition.

D. Organization of Report

Part II of the report provides a discussion of the SDV break scenario; the equipment qualification requirements of Section 5.3 of NUREG-0803; and a description of the Susquehanna specific break environment and how it applies to equipment qualification. Part III presents the preferred CSBD path as discussed above in Section I-C. Part IV is devoted to the component justification for interim operation, including a discussion of criteria employed in the evaluations. In Part IV, Conclusion, the preferred CSBD path is reviewed to determine whether all required components in the path are qualified or justified. If this goal is attained, it can be concluded that SSES Unit 1 can be operated safely pending qualification of equipment, if required, following NRC review of the BWR Owner's Group PRA.

II. NUREG-0803 Discussion

A. NUREG-0803 Break Scenario

NUREG-0803 postulates a break or leak to exist in the SDV piping during a reactor scram. This would result in the release of water and steam at 212°F into the reactor building at a maximum flow rate of 550 gpm, and is postulated to result in 100% relative humidity in the reactor building. The principal means of isolating this break would be to close the scram exhaust valves which are located on the Hydraulic Control Units (HCU); however, this is dependent upon the ability to reset scram, which cannot be absolutely ensured immediately following the scram. Therefore, a rupture of the SDV could result in an unisolable break outside of primary containment, which is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located. Consequently, NUREG-0803 provides guidance to ensure pipe integrity, detection capability, mitigation capability and qualification of the emergency equipment to the expected environment.

B. Section 5.3 Requirements

Section 5.3 of NUREG-0803 summarizes the equipment qualification requirements for components used to mitigate and/or identify an SDV break. There requirements are listed as follows:

(1) Identify the equipment that would be used to detect a break and/or leak in the SDV system and include the qualification of this equipment in the NRC's ongoing EQ program to show that it would perform the identification function. (This equipment and its environmental qualification is listed in Appendix B of this report).

(2) Identify the equipment needed to mitigate an unisolable break in the SDV system and include the qualification of this equipment in the NRC's ongoing EQ program to show that it would perform the mitigation function, paying particular attention to the guidance summarized in Table 5.1 of NUREG-0803. (This equipment and its environmental qualification is listed in Appendix D of this report).

When providing the qualification of this equipment, qualification for service at a given temperature and humidity shall mean that the equipment is capable of remaining on standby and of being energized and operated in the presence of the actual SDV break environmental transient (including specified level of adverse environment as a maximum condition) without any immediate or subsequent loss of required SDV break-mitigating capability. The SDV break environmental profile assumed for qualification purposes shall be prescribed as to temperature and humidity levels and time duration at each level, and shall be a conservative replica of the postulated transient.

(3) For any equipment required for identification and/or mitigation that is not qualified for service at 212°F and 100% humidity, provide a schedule for defining the plant-specific SDV break environment and a commitment to qualify the equipment in accordance with the NRC's ongoing EQ program.

For item 3 above, it is PP&L's intent to use the Susquehanna specific break environment for equipment qualification consideration, and not the 212°F value listed in item 3. A more detailed discussion of the Susquehanna break environment is given in Section II-C of this report.

C. Susquehanna Specific Break Environment

PP&L has performed a calculation to determine the temperature, pressure, radiation, and humidity in the reactor building following a break in the SDV. This calculation was performed as part of PP&L's response to NUREG-0803. The results for temperature and pressure are shown in Appendix A in the form of curves. The temperatures indicated these curves will be the ones used for equipment qualification evaluation purposes, since they represent the Susquehanna specific environments. Additionally, the only area effected by the SDV break is the Reactor Building outside of Primary Containment. The humidity is assumed to be 100% and the radiation effects on equipment was found to be negligible.

After examining the results of these calculations, several conclusions were made regarding the environment. First of all, all areas of the reactor building, with the exception of the immediate break area, reached their maximum temperatures in approximately 20 minutes. The temperature increase was gradual and the peak temperature lasted for a short duration (approx. 10 minutes) before the temperature gradually decreased due to depressurization of the vessel. The floor on which the SDV is located reached the highest temperature and had the longest duration. However the maximum temperature of 198°F is only attained for approx. 10 minutes before the temperature begins to decrease such that 40 minutes after peaking, the temperature has decreased to 145°F.

The worst effected areas of the plant are the 714' elevation (where the SDV is located) and the 749' elevation. These floors have motor control centers and switchgear which are critical to the operation of the Emergency Core Cooling Systems (ECCS) at Susquehanna.

III. Cold Shutdown and Break Detection (CSBD) Path

A. Break Detection Components

Section 5.3 of NUREG-0803 requires the licensee and applicants to identify equipment which would be used to detect a break in the SDV system. Additionally, the qualification of this equipment, must be addressed to assure its availability during the SDV break.

The equipment which would be most effective in detecting a break or leak in the SDV piping system consists of the CRD area radiation monitors, the CRD temperature monitors, and the Rod Position Indication System (RPIS). The combination of these signal sources will enable a correct event diagnosis by the operator to ensure correct operator action in response to a postulated break in the SDV piping system. Smaller pipe breaks and valve leakage from the CRD system can be detected from increased area radiation levels accompanied by a sustained trend in high CRD temperatures.

CRD Area Radiation Monitors: Area radiation monitors RIT-13705 and RIT-13706 provide an early indication of a leak in the SDV piping system. Radiation levels expected following a SDV system leak would activate a common reactor building high radiation alarm in the main control room. Radiation levels for specific areas are recorded in the main control room and are also indicated on the Area Radiation Monitor Cabinet.

However, these monitors are not in the EQ program and will be subjected to the break environment.

CRD Temperatures: A leak of any size in the SDV piping system will tend to increase the flow of high temperature reactor coolant past the CRD seals, especially where those seals are worn. Temperature elements mounted in the RPIS probes provide indication of SDV leakage by detecting a temperature increase in the water flowing past the CRD assemblies. High CRD temperatures are alarmed in the main control room. All CRD temperatures are recorded on a multipoint recorder at the CRD and RPV Temperature Recorder Panel. The recorder has a scan rate of one point every 5 seconds, which provides a complete survey of all 185 CRD temperature in less than 15-1/2 minutes.

This instrumentation will not be subjected to the SDV break environment since it is located in the main control room and inside the primary containment. Therefore, it will be available for use in the CSBD path.

Rod Position Indication: A large pipe break in the SDV piping system will provide a differential pressure across the control rod drives to force many rods into the overtravel position, similar to the position following a scram. At Susquehanna, the control rod overtravel position is not recognized by the RPIS, which will produce a blank display on the Full Core Display Module. Except for a loss of power to the RPIS, the blank control rod position display is unique to the overtravel position. This lack of position display will be accompanied by rod overtravel and rod drift annunciator and computer alarms.

This instrumentation will not be subjected to the SDV break environment since it is located in the main control room and inside primary containment. Therefore, it will be available for use in the CSBD path.

In summary, PP&L believes it is possible to provide sufficient leak detection without use of the area radiation monitors. This is based on the availability of CRD Temperature indication on all CRD's within 15-1/2 minutes; use of the Rod Position Indication System and precautions in the operating procedures to make the operator aware of a break in the SDV if the indications from the instruments described above are present following a scram.

B. Mitigation Components

Appendix C provides a list of all those components which may be used to mitigate a break in the SDV.

This list includes instruments, motor operated valves, pump motors, unit coolers, and power supplies. The systems which may be used are the following:

- Automatic Depressurization System (ADS): the manual function of this system would be used to depressurize the Reactor Vessel to allow use of the low pressure coolant supply systems.

- Residual Heat Removal System (RHR): This system would provide low pressure coolant injection (LPCI) and suppression pool cooling to provide for long term cooling.

- Core Spray System (CS): This system would provide low pressure cooling to the vessel for cold shutdown purposes.

- RHR Service Water System: This system is needed to support the RHR system.

- Emergency Service Water System: This system is required to support the RHR and CS systems.

- Condensate System: This system would be available to provide low pressure coolant to the reactor vessel.

The environmental qualification and the break environment of these components is given in Appendix D. From this appendix it can be determined which components are not qualified to the break environment. A more detailed discussion of which components are not qualified will be provided in Section IV of this report.

C. CSBD Path Selection

1. Background

The purpose of this section is to show that one path to cold shutdown and a means of break detection exists which uses equipment which is qualified for the break environment, or for which an acceptable justification for use exists. As previously mentioned in Section I-B of this report, only those components failures which are a direct result of the SDV break will be assumed to occur. Susquehanna Unit 1 will only comprise 2.3% of the Total PJM generating capacity when operating at full power (Unit 2 is not considered since it will not be in operation prior to Unit 1 1st refueling outage). Therefore the loss of Susquehanna Unit 1 should not adversely effect the grid and offsite power should remain available for use at SSES. The following subsections will describe the minimum CSBD based on the above criteria.

2. Cold Shutdown Path

Normally, in order to provide Extended Core Cooling RHR and ADS (in manual) would be used; while for Initial Core Cooling RHR, CS, and ADS would be adequate. However, from Appendix B it can be seen that RHR and CS would be unavailable due to the power supplies for the pumps being unqualified for the SDV break environment. But, based on the premise that no failure beyond the SDV break and its direct consequences occurs, the condensate system would be available to provide Extended Core Cooling. Since this system is normally used to supply makeup to the reactor vessel it is capable to supply more than enough flow to provide core cooling and replace the water lost through the SDV break. The four condensate pumps at Susquehanna are electrically driven pumps and are capable of supplying 8,250 gpm each (or 33% system capacity each) at 570 psig (at pump discharge). Additionally, all components of the condensate system are located outside of the SDV break environment and therefore are not required to be environmentally qualified for the SDV break. However, the feedwater injection valves through which the condensate must pass are subject to the break environment, but are qualified to the break conditions. The Automatic Depressurization System (ADS) also has its components located outside of the break environment and therefore is not affected by this environment (the ADS Components are located within Primary Containment and are qualified to a much harsher environment than the SDV break).

Based on the above information the described Shutdown Path is adequate.

3. Support System Identification

In order to use condensate as the mitigation path for the SDV break, the following support or auxiliary systems are required to be operational:

- Service water system
- Turbine Building Closed Cooling Water System
- The Main Condenser and associated equipment

Since all the above systems are located outside the break environment and would not fail as a direct consequence of the SDV break, it is assumed that these systems will be available and their qualification need not be considered.

In addition to the above, the following support System would be required to ensure operation of the RHR and Core Spray Systems in the event these systems are used:

- RHR Service Water
- Emergency Service Water

All components for these systems are located outside of the break environment except for the inlet and outlet valves for the RHR Heat Exchanger. However, as noted in Appendix D, these valves are qualified for the break environment and these receive their power supply from break environment qualified MCC's. Therefore, from the above information the support systems will be available to assist in mitigating the break.

4. Required CSBD Components

A listing of the required CSBD Components is provided in Appendix D. This list identifies the component, its power supply and its location so the qualification of this equipment can be addressed.

IV. Component Justification

All mitigation/identification components which are located in the SDV break environment and are not environmentally qualified to the break environment before have been evaluated to ascertain wether the SSES can be safely operated.

A. Criteria for justifications

The criteria for justification has been previously identified in PP&L's "Environmental Qualification Justification for Interim Operation for the Susquehanna Steam Electric Station," which was transmitted to the NRC via letter PLA-1084, dated May 7, 1982. However, since the SDV break was not considered in the design basis accidents for any nuclear power plant, it is necessary to make some clarifications to the criteria.

1) Any component which is used to mitigate an SDV break, whether it is Class 1E or not, must be justified to operate in the SDV break environment that it sees.

2) If a component which is located outside the SDV environmentis used to mitigate the break, it need not be environmentally qualified for the break.

3) NUREG-0803 does not postulate any failures other than the SDV break and its direct consequence to occur concurrent with the SDV break.

B. Detailed Justifications

The detailed component justifications appear on the forms found in Appendix F of this report. These forms give the basic information on each component which is not qualified to the break environment and is located in the environment.

The M/I components which are not presently qualified to the SDV break environment are motor control centers and switchgear. All others mitigation Components (MOV's, pump motors, unit coolers, and instrumentation) are currently in the EQ program and qualified to the break environment, or it is not located in an area which is subject to the break environment. The qualification of this equipment and its break environment can be found in Appendix D.

In the case of the motor control centers (MCC) and switchgear, they do not provide power to Components on the CSBD path and therefore failure of these MCC's will not effect the plant ability to shutdown, using the CSBD path described in Section III-C-1.

V. Conclusion

Interim operation of the Susquehanna Steam Electric Station Unit 1 without fully qualified mitigation/identification equipment for the SDV break environment has been justified. First, a single path to cold shutdown has been established which is qualified or for which an acceptable justification exists. Additionally, PP&L has participated in a BWR Owner's Group effort to perform a PRA based on plant specific data to determine the probability of a SDV break occurring. The results of this PRA will eliminate the need for additional equipment qualification. This PRA was submitted to the NRC on August 30, 1982. Therefore, based on the above information, PP&L believes that interim operation of SSES Unit 1 is justified.

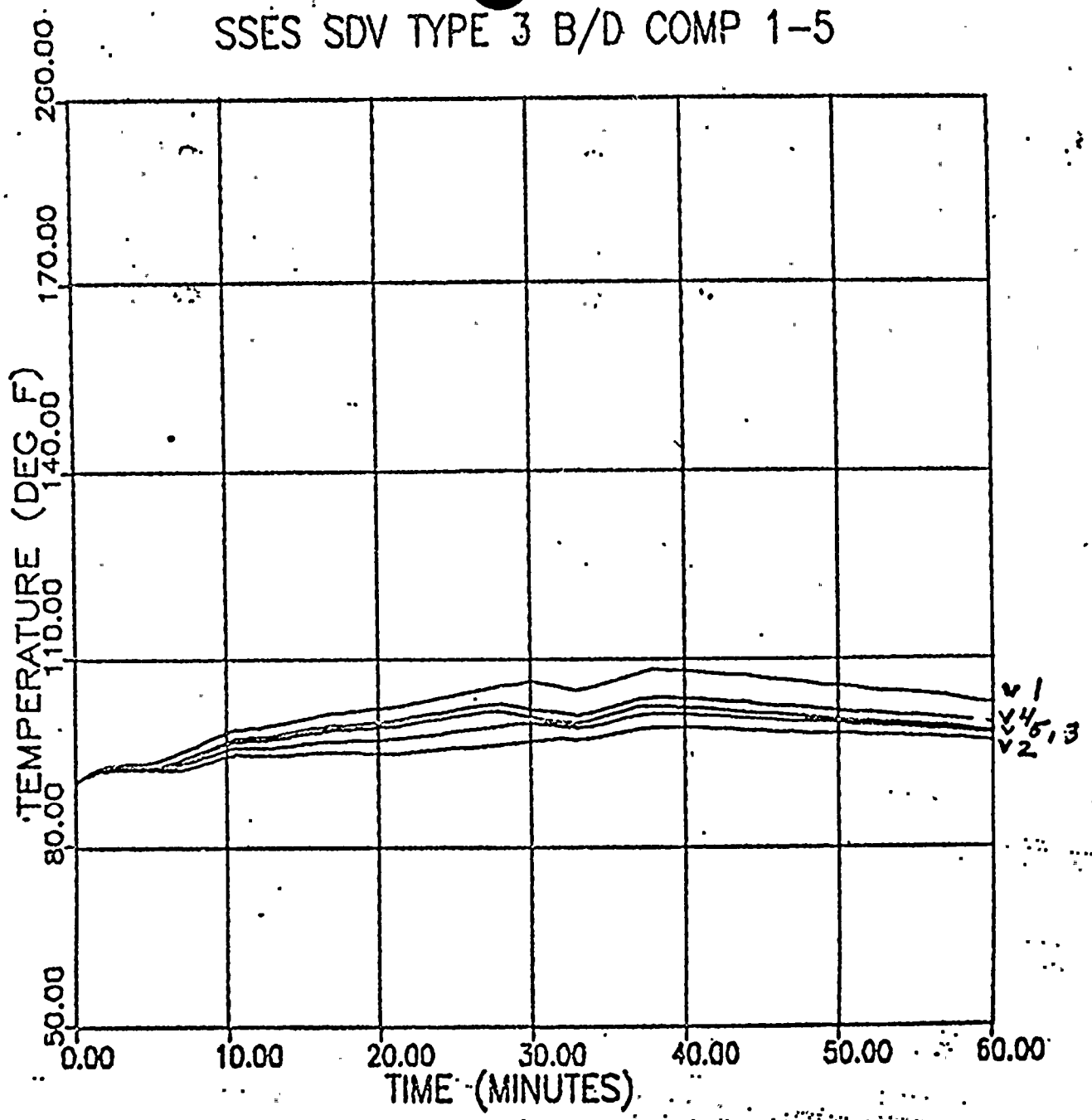
APPENDIX A

SUSQUEHANNA STEAM ELECTRIC STATION SPECIFIC

SDV BREAK ENVIRONMENT

(TEMPERATURE & PRESSURE CURVES)

Compartment No.	Room No.
V - 1	I-13, I-103 RHR Pump Room, El. 646 & 670, North Hal
- 2	I-14, I-104 RHR Pump Room, El. 646 & 670, South Hal
- 3	I-15 LRW Sump Room (El. 646)
- 4	I-17 Core Spray Pump Room, El 646, South.
- 5	I-10 Core Spray Pump Room, El. 646, North.
- 6	I-11, I-106 HPCI Pump Room & Pipe Area El. 646.
- 7	I-12, I-07 RCIC Pump Room & Pipe Area El. 646 & 670
- 8	I-102 Core Spray Pump, North El. 670
- 9	I-105 Core Spray Pump, South El. 670
-10	I-200, I-201 Equip. Removal Areas. El. 683
-11	I-202, I-204, I-205 Equip Areas El. 683'
-12	I-203 Heat Exchanger & Pump Room El. 683
-13	I-401, I-408 North HCU Area & Gas Accumulator Room (El 719'-1")
-14	I-403 TIP Room El.719'-1"
-15	I-406, I-407 Emergency Switch Gear Rooms El. 719'-1"
-16	I-410 Control Rod Drive Room (El. 719'-1")
-17	I-411 Main Steam Pipeway (El. 719'-1")
-18	I-501, I-502, I-503 Penetr. Room, Clean-up Recirc. Pump Room (El. 749)
-19	I-504, I-505 Heat Ex. Room El. 749
-20	I-500, I-506, I-508, I-511, I-513, I-514, I-515 General Floor Area El. 749
-21	I-517, I-510 Load Center Rooms El. 749'
-22	Supply Header
-23	Exhaust Header
-24	Zone 3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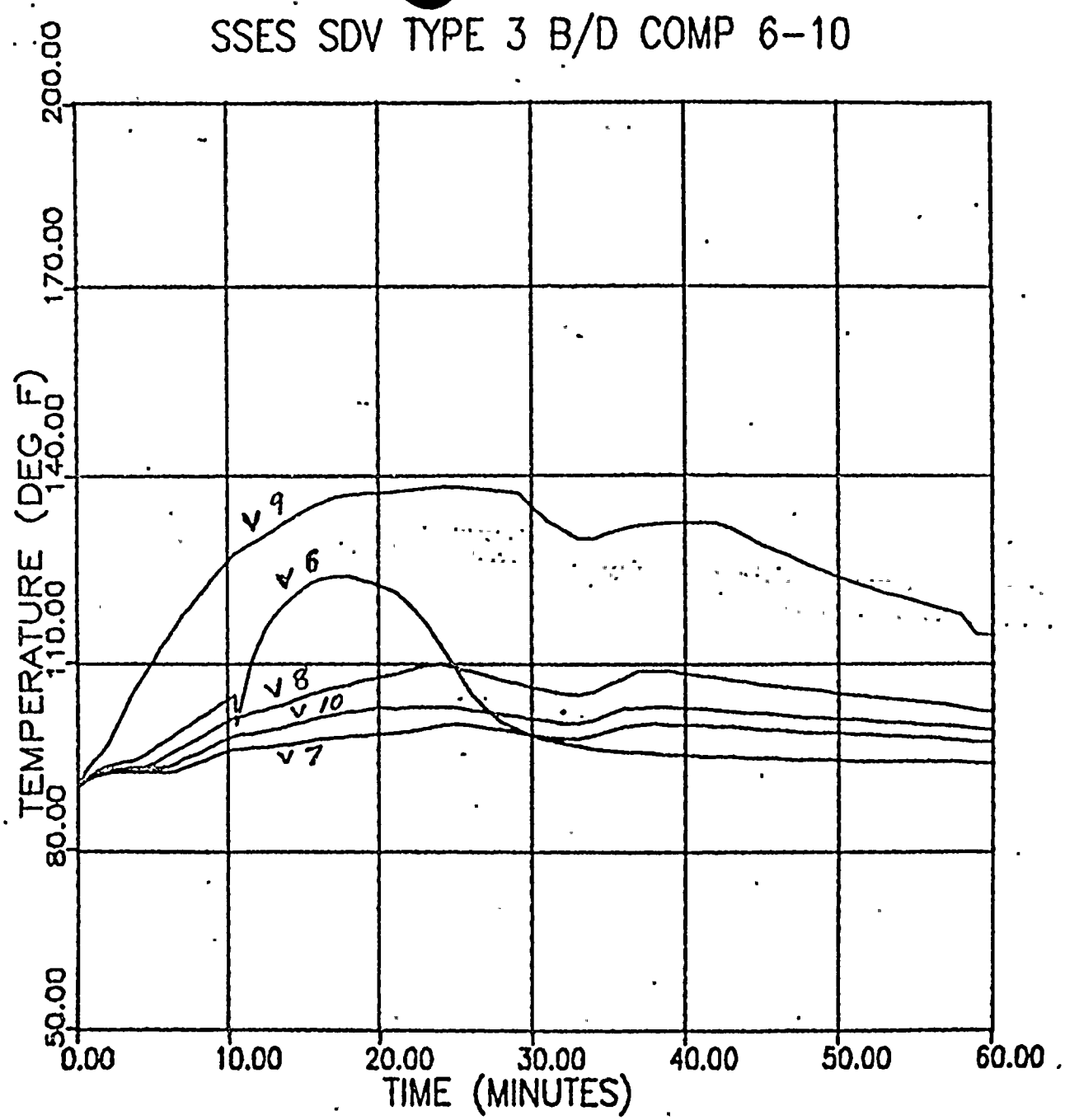
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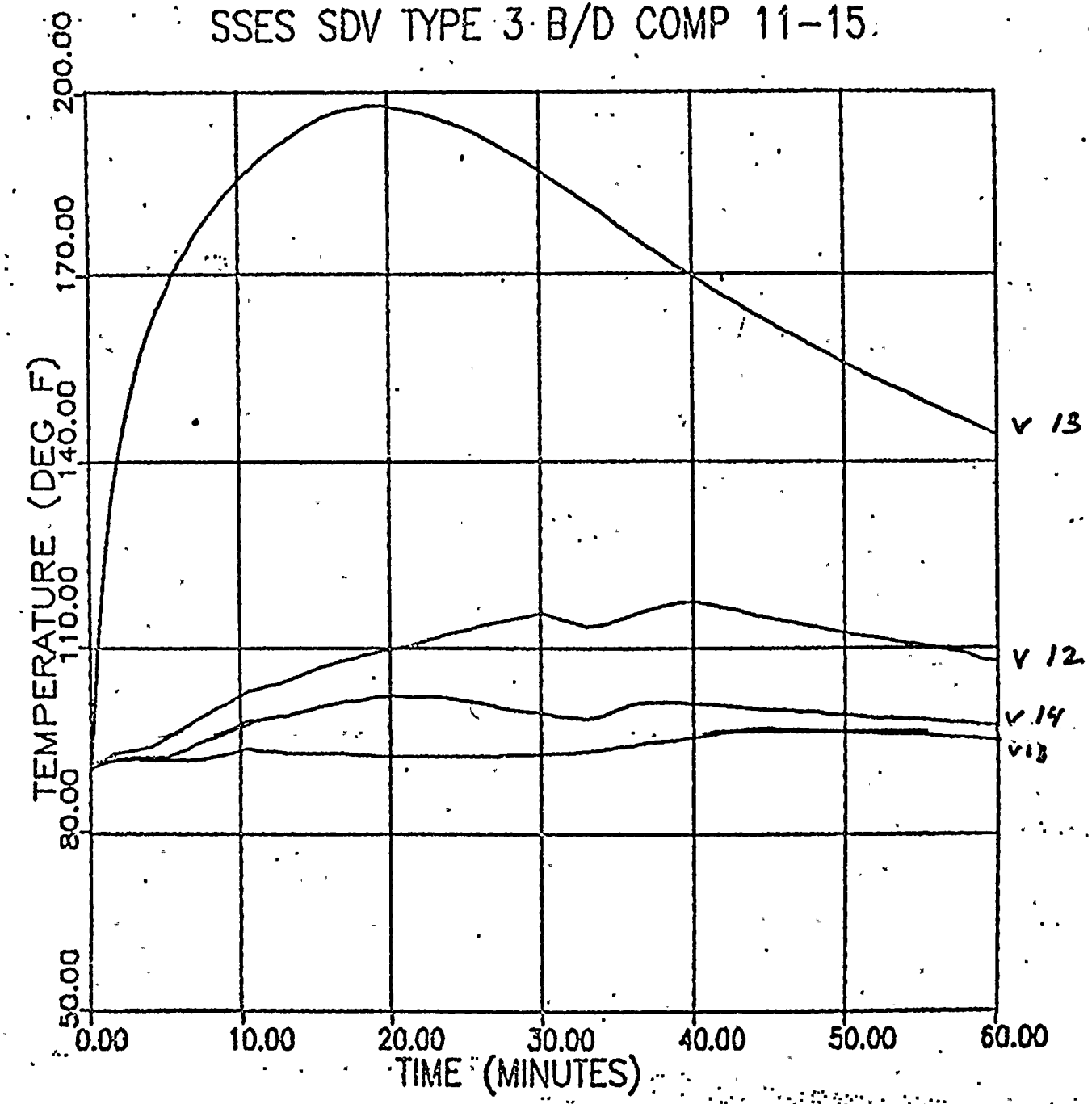
V 6
V 8
V 10
V 7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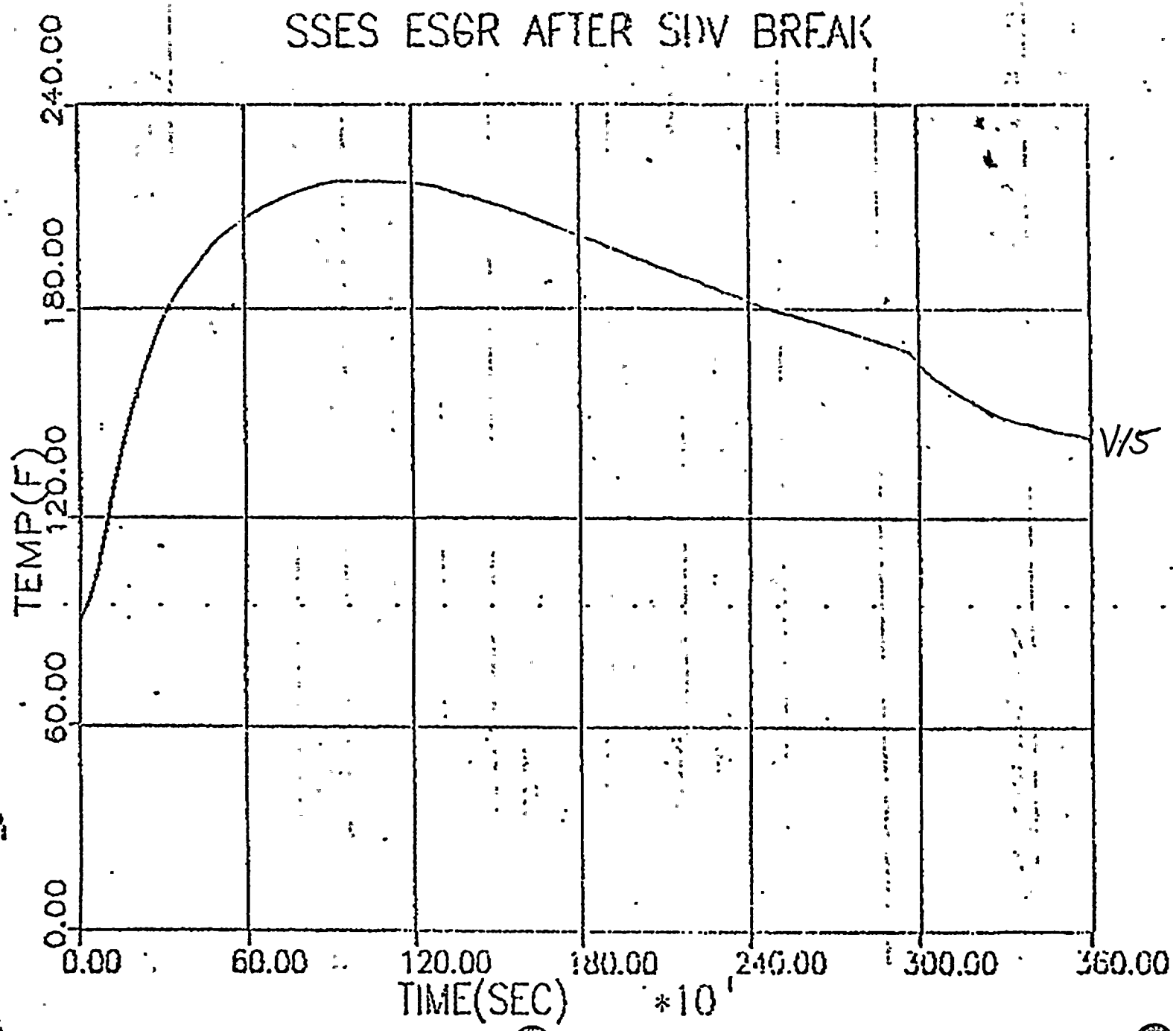
V 12
V 14
V 18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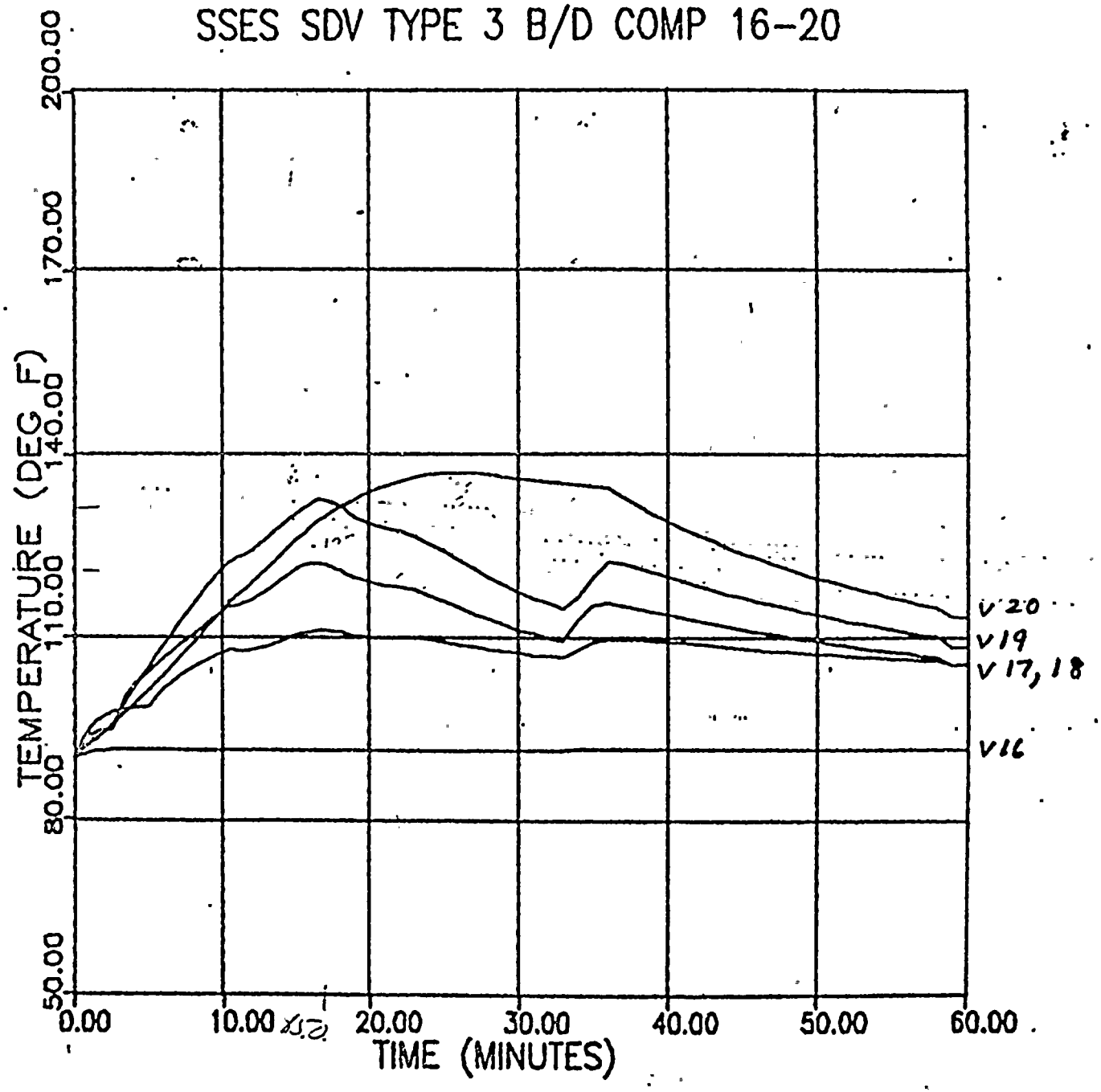
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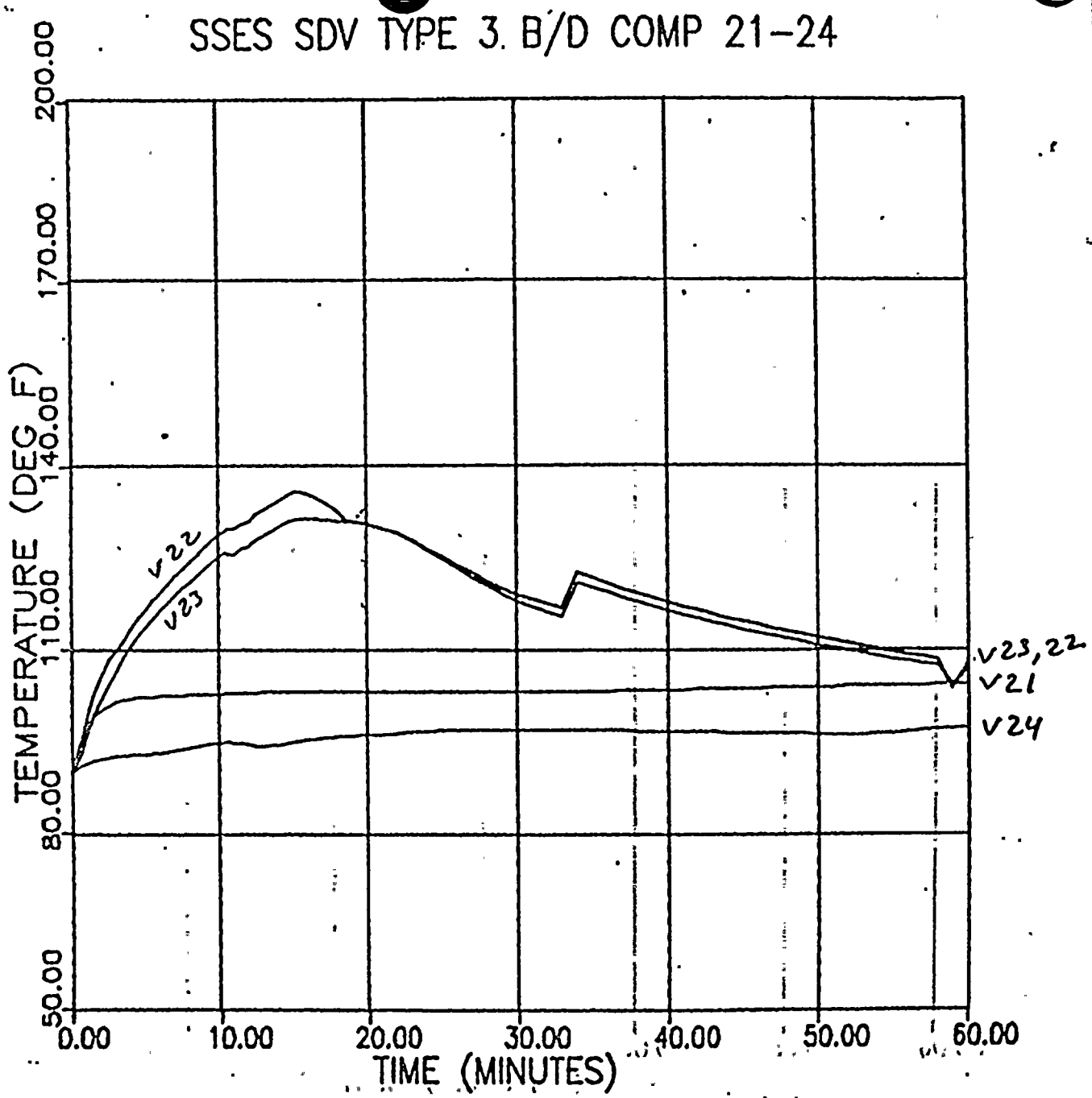
V19
V17, 18
V16

0273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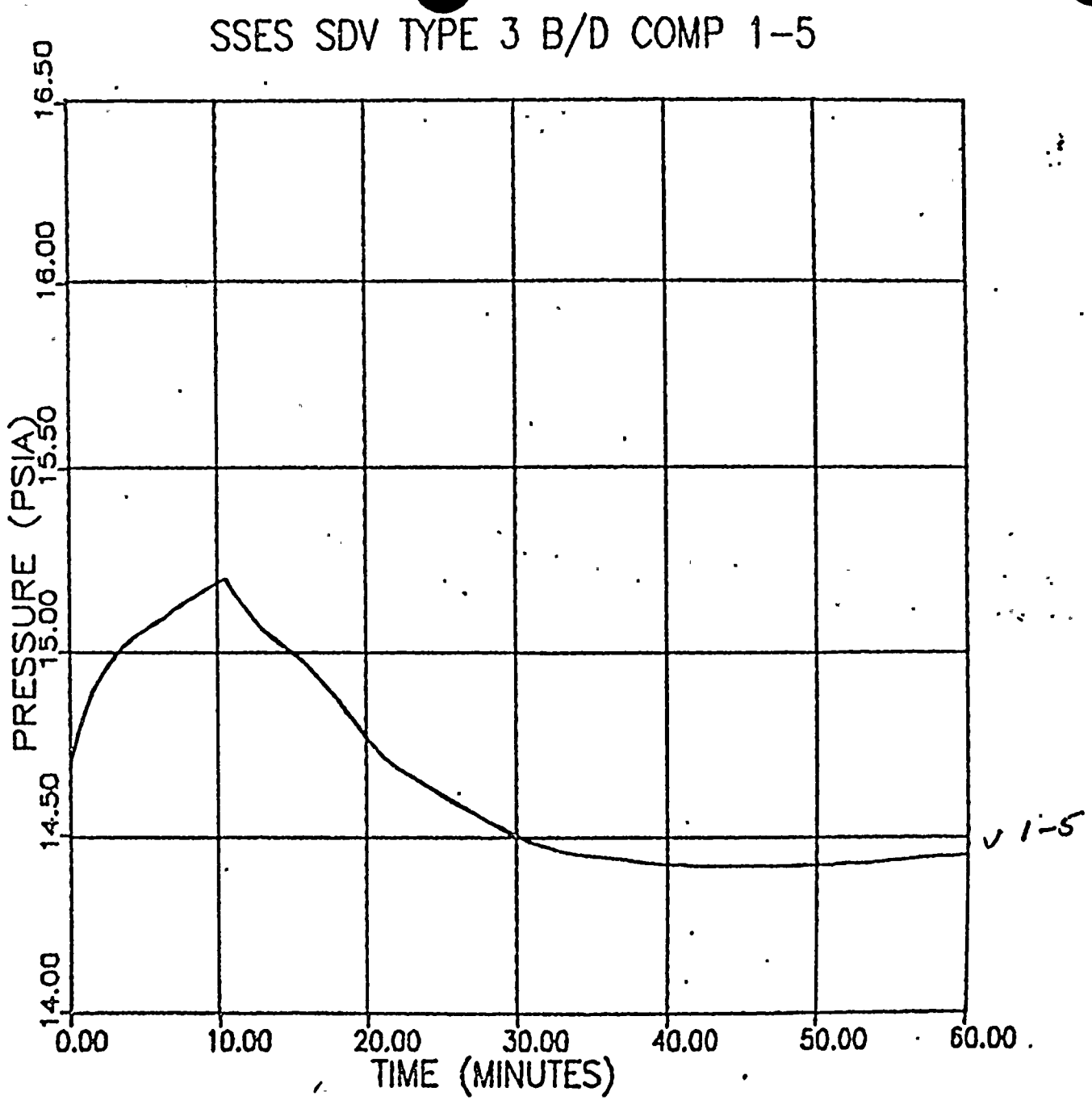
V23
V23,22
V21
V24

0274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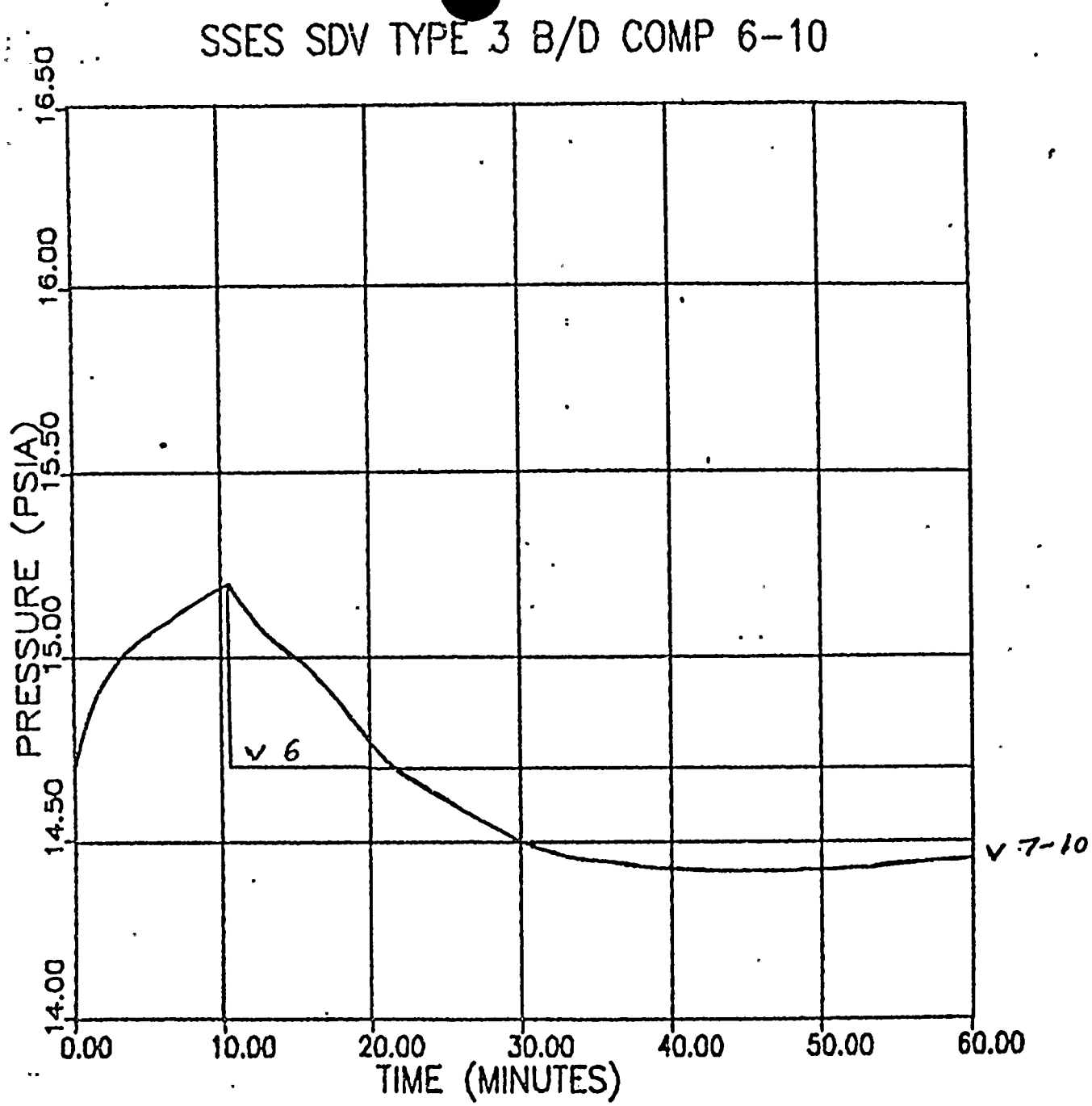
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V 7-10

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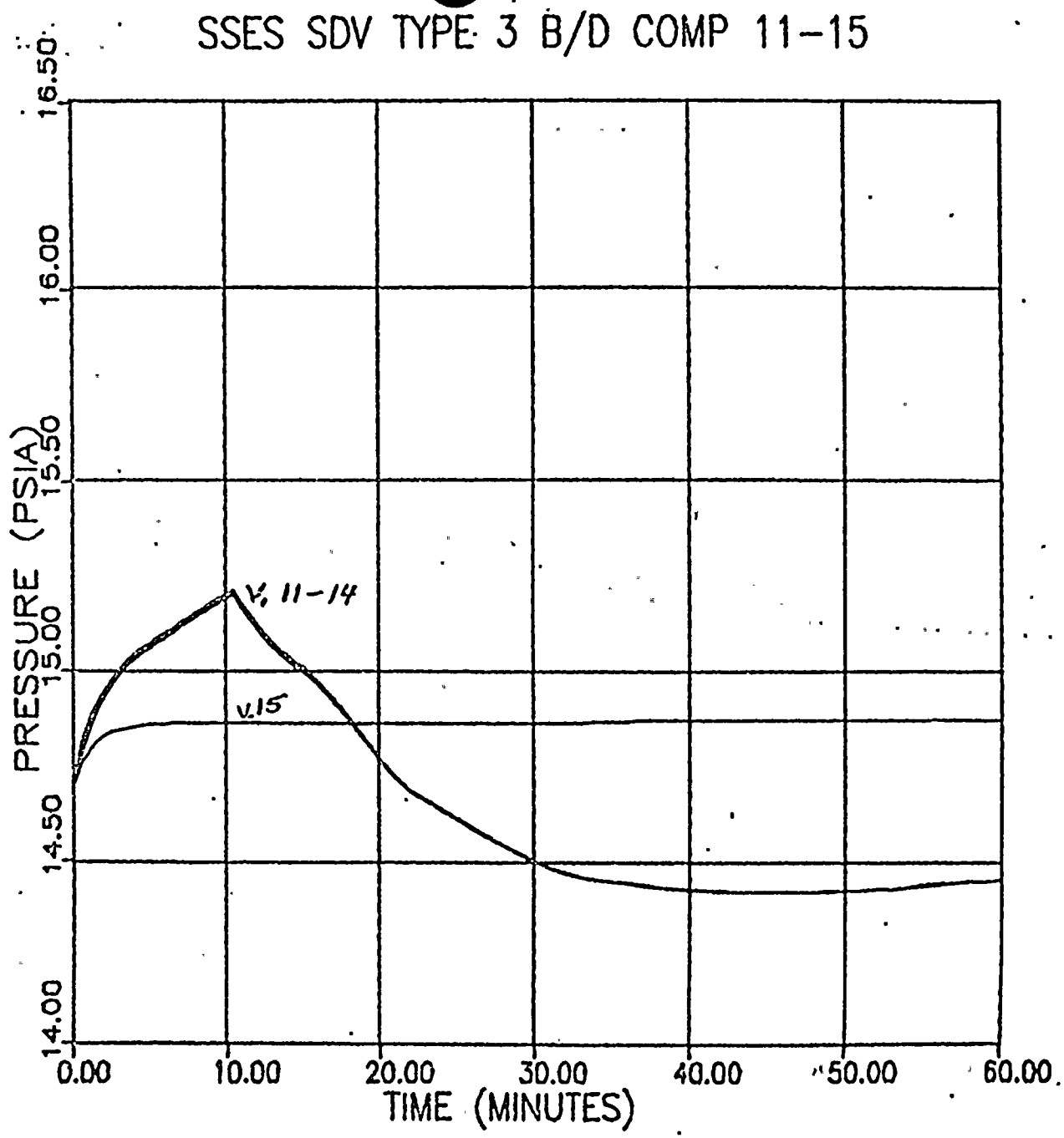
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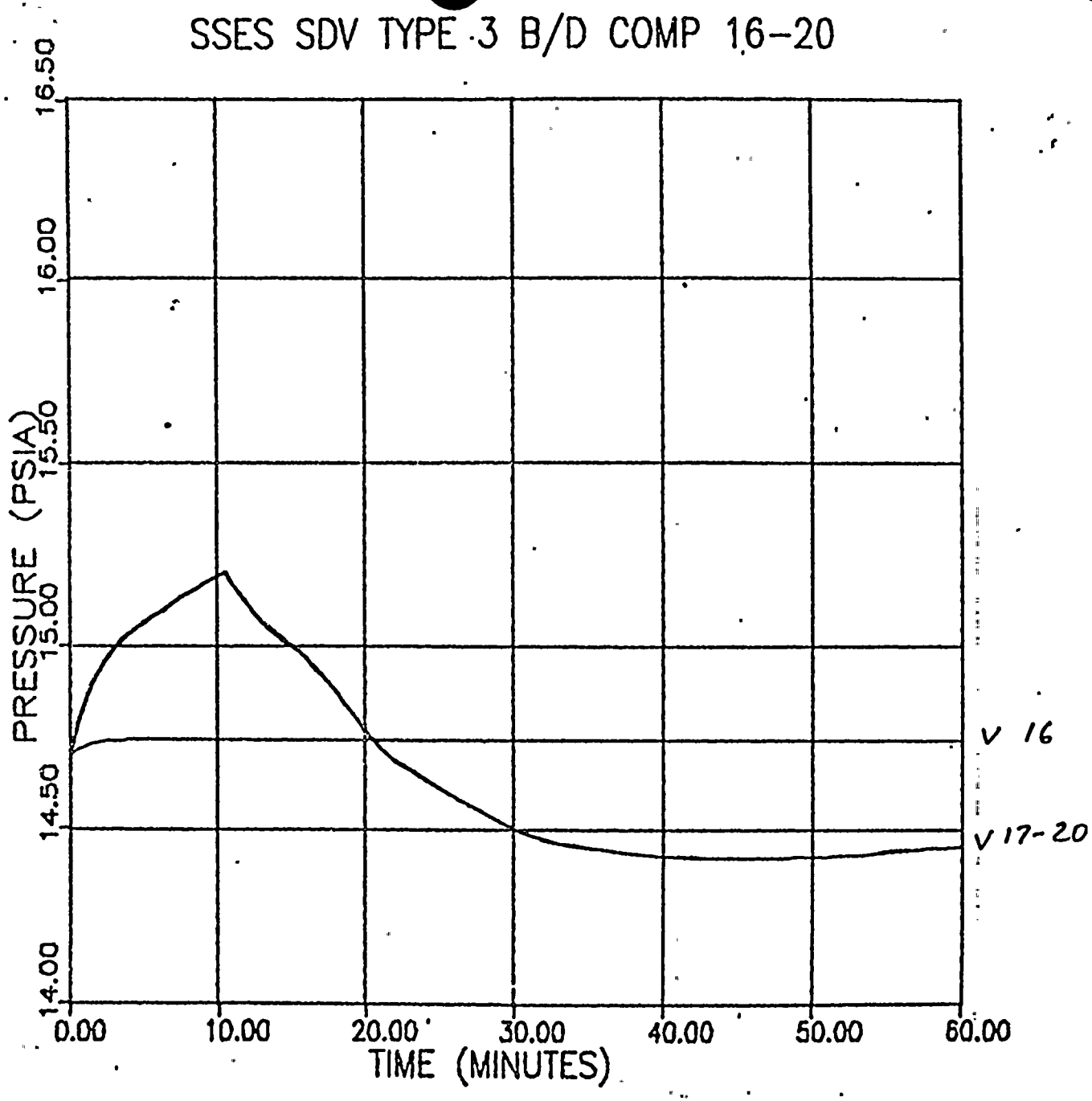
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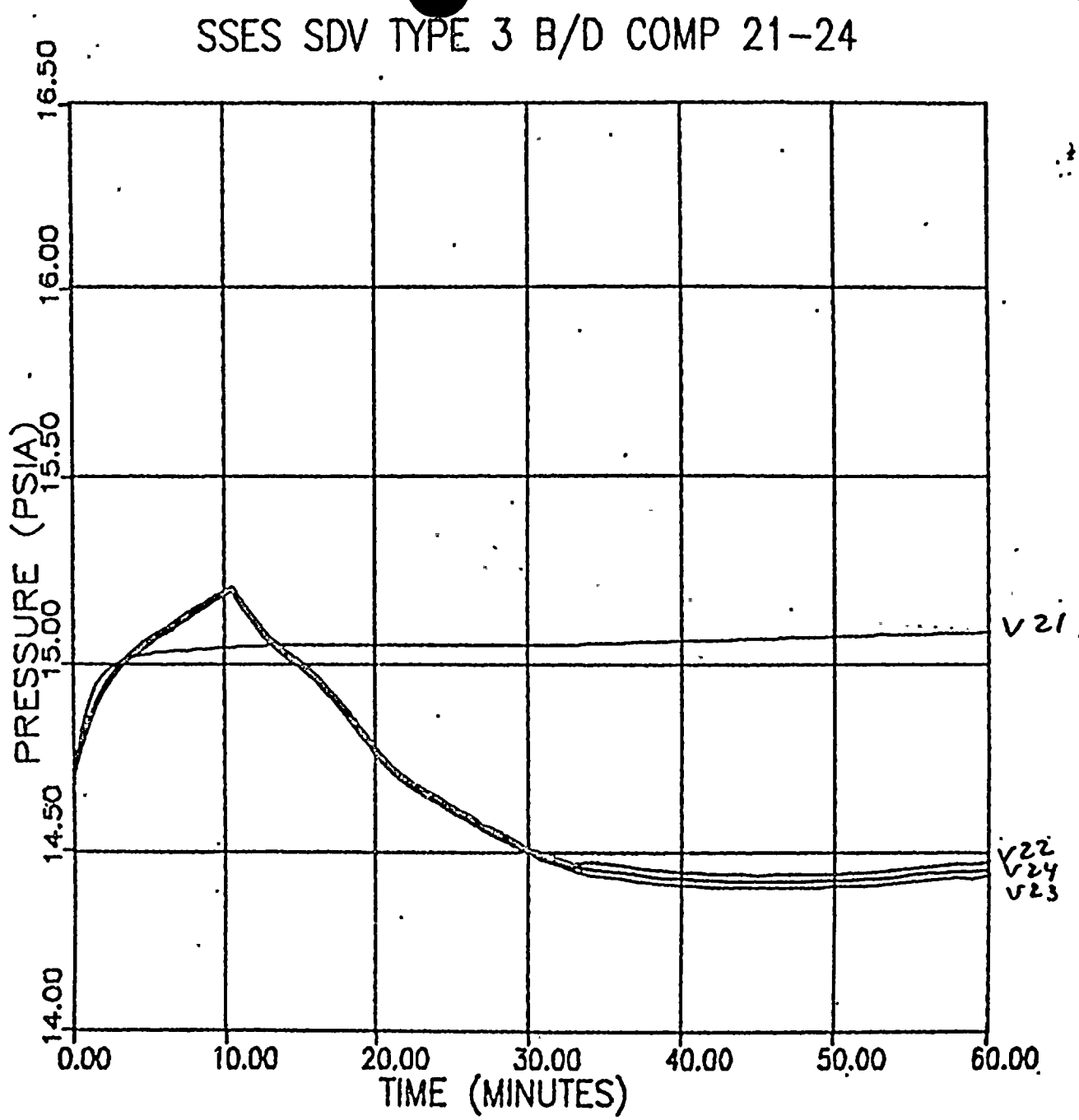
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V22
V24
V23

0278

APPENDIX B

BWR OWNER'S GROUP

PRA

FOR NUREG-0803

NEDO-22209
82NED088
CLASS I
August 1982

DRAFT COPY

ANALYSIS OF SCRAM DISCHARGE VOLUME SYSTEM PIPING INTEGRITY

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Approved by: for R.J. Brandon
R.J. Brandon, Manager
Nuclear Services Engineering Operation

Approved by: J.F. Quirk
J.F. Quirk, Manager
BWR Systems Licensing

Approved by: _____
J.F. Schilder, Manager
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0.3 ABSTRACT

Analyses of the Boiling Water Reactor (BWR) scram system piping integrity have been performed. The purpose of these analyses is to determine the probability of a loss of SDV piping integrity and to evaluate the contribution of such a loss to a core melt.

The likelihood of a loss of piping integrity was calculated based on a consideration of pipe length, scram frequency and vent and drain valve reliability. Conservative values for the key input values were selected based on BWR plant data and on generic reliability data. Pipe break probabilities were estimated based on the experience data used in the Reactor Safety Study and on a fracture mechanics analysis of the piping system.

The results of these analyses show that the probability of an unisolatable loss of scram system piping integrity for an average plant is 3×10^{-7} per plant year. The probability of core damage resulting from a loss of SDV pipe integrity is approximately 4×10^{-11} events per reactor year. This is significantly below the proposed NRC safety goal for core melt events of 10^{-4} per plant year. Consequently, the probability of a loss of scram system piping integrity leading to core damage is sufficiently low to preclude the necessity of qualification or design modifications of equipment required to detect and/or mitigate the consequences of such an integrity loss.

1. Introduction

1.1 Background

In August 1981, the NRC issued the results of a generic review of pipe breaks in the BWR scram system piping in NUREG-0803 ''Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping''. (Ref. 1). The NRC concluded that for Mark I and Mark II containment plants the scram system piping is acceptable provided that steps be taken to: (1) ensure the piping integrity, (2) mitigate the consequences of a scram discharge volume (SDV) break, and (3) environmentally qualify the equipment required to detect and/or mitigate the consequences of the break.

The need for mitigation measures and equipment qualification was predicated on an estimated probability of SDV pipe break being sufficiently high that it could not be dismissed. Implicit in this approach is the argument that if the probability of a break in the SDV piping is sufficiently low, then consideration need not be given to mitigation features and equipment qualification for that particular break.

Using a defect rate of 3×10^{-7} per foot of pipe per year and an estimated SDV piping length of 2500 ft, the NRC calculated an SDV failure rate of 10^{-4} per plant year. It noted that this value is extremely conservative since the SDV would be under load less than 1% of the time.

An earlier report, NEDO-24342, ''GE Evaluation In Response To NRC Request Regarding BWR Scram System Pipe Breaks'' (Ref. 2) used WASH-1400 (Ref. 3) values to evaluate the SDV break probability. It calculated the ratio of the SDV pipe length to the LOCA sensitive piping length and took into consideration the diameter of the pipes. (LOCA sensitive piping is that piping inside the containment that would result in a loss of reactor coolant in case of a break.) This approach yielded a break probability of 3×10^{-6}/plant year taking into account the fraction of time the SDV piping is pressurized. Both NEDO-24342 and NUREG 0803 used estimated conservative generic plant data.

1.2 Purpose

It is the purpose of this report to perform a more detailed analysis of the failure probability of the SDV taking into account plant specific data, in order to demonstrate that an SDV failure resulting in a substantial leak which could threaten equipment required to detect and/or mitigate the leak is not a credible event.

Three different approaches will be used:

1) the NEDO-24342 approach

2) the NUREG-0803 approach

3) the fracture mechanics approach

The last approach evaluates break probabilities by analyzing the mechanism of crack growth while under repeated stress.

2.0 Analysis

2.1 Description of SDV System

The scram discharge system receives the water exhausted from the control rod drives (CRD) during a reactor scram. For a short time during and following each reactor scram, it contains reactor coolant at full reactor pressure. This section briefly describes the fundamentals of operation of the system.

The scram discharge system, which is depicted in Figure 2.1, consists of the CRD, the CRD withdraw lines, the scram discharge volume and the valves associated with the discharge volume.

During a scram, water from the volumes above the CRD pistons is discharged to the CRD withdraw lines. It flows through the scram valves to the scram discharge volume. The scram discharge volume vent and drain valves are open during normal operation, and close automatically on receipt of a scram signal.

The discharge volume partially fills with the water discharged from the CRDs. Upon completion of a reactor scram, with all control rods fully inserted, water leaking past the CRD seals from the reactor and water from the CRD pump continues to flow into the scram discharge volume. This flow continues until the pressure in the scram discharge volume is equal to the reactor pressure.

When the scram signal is reset by the operator, the scram valves close and the scram discharge volume vent and drain valves open. The scram discharge volume empties and returns to atmospheric pressure, configuring it for normal operation.

The scram valves and the scram discharge volume vent and drain valves are diaphragm actuated. These valves are designed to move into their scram positions when air pressure is removed. Motive air from the reactor building instrument air system is supplied to these valves via solenoid-operated pilot valves actuated by the reactor protection system. Two normally open manual isolation valves are provided at each hydraulic control unit to isolate the scram discharge volume from the CRD.

The system, because of its simple design, provides a high reliability to scram: and because the valves assume their scram positions when air pressure is removed, the reactor will be shut down automatically if the air supply becomes unavailable.

Figure 2.2 shows additional details of the scram discharge volume itself. To comply with the SDV Safety Evaluation Report (Ref. 4) all SDV have or will have two vent valves in series and two drain valves in series. Also, some systems currently have a relief valve. Table 2.1 summarizes the details of each plant including pipe lengths as a function of diameter, design code used, number and types of joints and scram history. The piping system which is of interest for this study is that portion which extends from the check valves upstream of the SDV header up to and including the vent and drain valve piping.

2.2 Fault Tree Diagram

2.2.1 General Description

Figure 2.3 shows a fault tree diagram for the SDV system shown in Figure 2.2. The top event consists of any violation of the integrity of the SDV including pipe breaks and valve malfunctions that would result in water spilling into the reactor building. Two events need to occur; the SDV integrity must be breached and the reactor must be scrammed (i.e., the SDV and associated piping must be pressurized)

There are several ways that the SDV integrity can be breached: (1) a break in the pipe, (2) the relief valve fails open, and (3) two drain and/or two vent valves are stuck open. The relief, drain and vent valves are typically all piped to sumps in the basement. Depending on the size of the sump(s) and capacity of the sump pump(s), stuck open valves during a scram that are not or cannot be reset could lead to eventual overflow of the sump. For this reason, the stuck open valves are considered as a failure of SDV integrity. However, the consequences are expected to be considerably less significant than those for a break.

2.2.2 SDV Pipe Break Probability

2.2.2.1 Review of NEDO-24342 Approach

The SDV pipe break probability has been previously addressed in NEDO-24342 (Ref.2). NEDO-24342 followed the approach used in Appendix 3 of WASH-1400. It used the assessed break probability for a LOCA. However since the piping length for the SDV is different than the length of LOCA sensitive piping, the probabilities were modified by the ratio of SDV piping length to LOCA sensitive piping length. This approach resulted in a break probability of 3×10^{-4} per year assuming the SDV is constantly pressurized. It estimated that a reactor is scrammed (SDV pressurized) 1% of the time. Thus an overall break probability of 3×10^{-6}/plant year resulted.

2.2.2.2 Review of NUREG-0803 Approach

NUREG-0803 used a different approach than that used in NEDO-24342. It estimated an SDV piping length of 2500 ft and multiplied it by a failure rate of 3×10^{-7} per foot per year to obtain a break probability of 10^{-4} per plant per year. It also noted that the SDV is only pressurized 1% of the time but it did not factor it directly into the break probability. If it were included, the result would have been very similar to that of NEDO-24342.

2.2.2.3 Re-evaluation of Break Probability Using Plant Specific Data

2.2.2.3.1 Evaluation Procedure

Using plant specific data, the SDV break probability was reevaluated following both the NUREG-0803 and the NEDO-24342 approaches.

The plant specific data that are being considered are the actual piping diameters, lengths, and scram histories. Following NEDO-24342 the SDV piping was first grouped into three diameter sizes- $<2''$, $\geq 2''$ to $6''$ and $> 6''$. (See Table 2.1).

The ratio of these lengths to the length of LOCA sensitive piping of the same diameter grouping were evaluated. The total length of LOCA sensitive piping was taken to be 6000 ft (Ref. 5). Following WASH 1400, the total length was equally apportioned among the three pipe groups. Thus each group consists of 2000 ft of pipe.

The median probabilities for a break in 2000 ft of LOCA sensitive piping from WASH 1400 are:

1/2'' to 2'' diameter	1×10^{-3}/plant year
2'' to 6'' diameter	3×10^{-4}/plant year
>6'' diameter	1×10^{-4}/plant year

Using these values and plant specific data from table 2.1 the probability of a break was evaluated.

The break probability was also evaluated using an approach similar to that in NUREG-0803. This involves multiplying the SDV pipe length by a defect rate of 3×10^{-7} per foot per year. (Ref. 3). The final break probability is evaluated by multiplying this preceding product by the fraction of time the plant is scrammed, (i.e., that SDV is pressurized) based on the scram history for that plant.

2.2.2.2.2 Discussion of Results

The SDV pipe break probability was evaluated for the ''average'' plant and for the ''limiting'' plant. The average plant refers to a plant having the average pipe lengths, number of scrams and scram duration from the data in Table 2.1. The limiting plant is defined as the plant with the longest pipe lengths, the largest number of scrams and longest average scram duration based on the data compiled in Table 2.1. The results appear in Table 2.2; the following observations can be made:

a) Both the NEDO-24342 and the NUREG-0803 approaches yield very similar results.

Since the WASH 1400 break probability numbers used in NEDO-24342 are in part derived from the number of defects per foot per year (Ref. 3), the similarity of the two results might have been anticipated.

b) The break probabilities are about two orders of magnitude lower than those obtained in NEDO-24342 and NUREG-0803

This results from the fact that plant specific data show that the SDV system is pressurized much less than the 1% assumed in the previous analyses. Table 2.2 indicates the fraction of time scrammed (i.e., pressurized) for the average and limiting plant. This is the biggest contributor to the reduction in the break probability.

c) The dominant contributor to the break probability are pipes of less than 2'' in diameter.

This is because most of the SDV piping length is small diameter piping; typically 70% or more is less than 1'' in diameter, with resulting low leakage flow rate. If the consequences of a small pipe break could be dismissed this would reduce the consequential pipe break probability by at least another factor of 10.

However, even including small pipes, the resulting break probability based on either the GE or NRC approaches is, on the average, less than 2×10^{-7} per plant year.

Note that no credit has been taken for installation examinations, the design code and piping class, the seismic class and inservice inspection. As indicated in Table 2.1, these factors are present in all plants and would further reduce the break probability.

2.2.2.4 Fracture Mechanics Approach

The two previous methods used to determine the break probabilities are based on accumulated experience. An alternate method is the fracture mechanics approach which examines the failure of pipes due to growth of crack-like defects that may be introduced into welds during fabrication of the pipe. (Ref. 6,7) This method will be used to support the results from the experience approaches.

The fracture mechanics approach is described in Reference 6 and has been applied in Reference 7 to analyze the probability of a pipe break in an SDV. It was found that the small pipes bound the large pipes in probability of failure. The small pipes are analyzed in this report following the method used in Reference 7, but using the SDV stress values from NEDO-24342 (Ref. 2).

The fracture mechanics approach investigates the probability of low-cycle fatigue causing through-wall crack propagation in the SDV piping system over the plant lifetime. This method assumes that piping failures occur due to the growth of defects introduced into welds during fabrication of the pipe. These initial defects are considered to be randomly distributed in both the number of defects and their size. The failure probability during a stress cycle equals the probability of a crack being larger than the critical crack size, given that a crack exists.

The stress levels assumed for this evaluation are the peak cyclic stresses in the SDV piping. The maximum stresses are (Ref. 2):

Pressure	1.5 Ksi
Temperature	1.2 Ksi
Total	2.7 Ksi

Deadweight stresses are not included because they do not contribute to fatigue. Seismic stresses are not accounted for because they contribute a small number of cycles. Typically only one operating basis earthquake can be expected during plant life ($p < 10^{-2}$/ry) and the probability of a safe-shutdown earthquake is less than 10^{-4} per reactor year. Water hammer effects on the SDV are not expected to be significant. Fast opening of the scram valve will result in a simple compression (Ref. 4) of the SDV since it is empty or near empty of water at the start of a scram. Opening of the drain or vent valves is also not expected to produce significant stresses since they drain into air filled pipes at atmospheric pressure. This will result in simple decompression of the SDV.

Intergranular stress corrosion cracking, as pointed out in NUREG-0803 is not expected to be a potential failure mechanism, because the SDV is pressurized for only a short period of time.

Scram frequencies of 9 (average) and 17 (maximum) per year are used (from Table 2.2). This amounts to 360 and 680 cycles over the plant life, respectively.

The initial crack distribution accounts for the probability that a crack exists and the size distribution of cracks given that a crack exists. The crack probability in a weld of volume, V, is Poisson distributed according to

$$P_c = 1 - e^{-VA} \qquad (1)$$

where:

A = crack existence frequency $10^{-4}/in^3$
V = $2\pi(ID)h^2$, $inch^3$
ID = Pipe ID, inch
h = Pipe thickness, inch

The size distribution of cracks, given that a crack exists, is distributed exponentially with a complementary cumulative distribution

$$P_{s/c} = 0 \qquad x > h$$

$$P_{s/c} = \frac{e^{-x/\lambda} - e^{-h/\lambda}}{1-e^{-h/\lambda}} \qquad 0 \leq x \leq h \qquad (2)$$

where λ = average crack size, inch, and h here represents the maximum crack size.

The SDV's undergo preservice proof testing. Positive results from this test insure that no cracks above a certain size, a_p, exist. (If they existed the pipe would fail during the proof test) Equation (2), thus, becomes:

$$P_{s/c} = 0 \qquad x>h$$

$$P_{s/c}(a>x) = \frac{e^{-x/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} \qquad 0 \leq x \leq a_p \qquad (3)$$

where:

a_p is the largest crack size that would survive proof testing.

Each stress cycle increases the size of the cracks. The crack growth rate per cycle for stainless steel is given by: (Ref. 7):

$$\frac{da}{dn} = 10^{-9}(\Delta K)^4$$

where:

$\frac{da}{dn}$ = crack growth rate, inches/cycle

ΔK = cycle stress intensity factor, $ksi\text{-}in^{1/2}$

$$= \Delta\sigma a^{1/2} \left(\frac{2 + C_1\alpha + C_2\alpha^2 + C_3\alpha^3 + C_4\alpha^4}{(1-\alpha)^{1/2}} \right)$$

$\alpha = \frac{a}{h}$ $\Delta\sigma$ = cyclic stress

$C_1 = -1.00250$ $C_3 = -6.21135$

$C_2 = 4.79463$ $C_4 = 1.79864$

The SDV consists of both stainless and carbon steel. The above relationship applies to stainless steel but it will be applied to carbon steel as well for conservatism.

The crack continues to grow until it reaches a critical size, a_c, at which point the pipe is assumed to fail. The critical crack size is given by (Ref. 7):

$$a_c = h\,(1-\sigma_{Lc}/\sigma_{cs})$$

where

σ_{Lc} = load controlled stress = $\sigma_p + \sigma_{dw}$

σ_p = stress due to pressure

σ_{dw} = stress due to deadweight

σ_{cs} = critical stress (flow stress)
= (yield strength + tensile strength)/2
= 45ksi for stainless and carbon steel (Ref.7)

To evaluate the pipe failure probability consider the tolerable initial crack size, $a_t(n)$. This represents an initial crack size that would just grow to the critical size after n stress cycles. The probability of failure within n cycles is then equal to the probability of having a crack larger than $a_t(n)$ at time zero. This is given by

$$P_{f(cond)}(n) = P[a > a_t(n)]$$

$$= \frac{e^{-a_t(n)/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} \qquad 0 \leq a_t(n) \leq a_p$$

$$= 0 \qquad \text{Otherwise}$$

The tolerable initial crack sizes, $a_t(n)$, can be evaluated using:

$$a_t(n) = a_t(n-1) - \frac{da}{dn}\Big|_{a = a_t(n-1)}$$

Finally, the unconditional average failure rate for the SDV system can be found using

$$\bar{P}_f = P_c \times P_{f\,(cond)} \times L/t$$

where L is the number of welds in the SDV,
t is the life of the plant and
$P_{f(cond)}$ is evaluated over the life of the plant.

This approach resulted in no failures for the aforementioned cyclic stresses (2.7 ksi) for both the average and maximum number of scrams cases. The reason for this is that the cyclic stresses are not sufficient to increase a crack from a_p (the proof test crack size) to the critical crack size, a_c. The minimum stresses that would accomplish this are ~6.5 ksi for 9 scrams/year and ~5.5 ksi for 17 scram/year. This is over twice the peak cyclic stress expected for a typical SDV. This result was obtained even with the use of the following conservative assumptions.

1) The influence of in-service inspection was ignored.

2) Only pre-service proof test was considered. In-service proof tests were ignored.

3) Stress intensity factors were conservatively estimated assuming all cracks to be fully circumferential.

4) The initial crack depth distribution for thick piping was used. This has a significant effect on the probability of having cracks greater than tolerable depth.

5) Upper bound estimate on fatigue crack growth characteristics were employed.

6) Conservative estimate of the flow stress was used.

7) All welds in the SDV system were assumed to be subjected to the maximum stress.

These fracture mechanics results support the outcome of the experience approaches which show that the probability of an SDV pipe failure is insignificant.

2.2.3 Probability of Stuck Open Valves

As pointed out in section 2.2, water from the SDV could spill onto the reactor building basement floor if the two drain valves or the two vent valves or the relief valve (if the plant has one) were to remain open after a scram that could not be reset. This event would not be as serious as a break since no water would be sprayed at the equipment. Typically instead, the water would simply flow to the sump. At this time the reactor building is assumed to be accessible, allowing personnel to close the manual SDV isolation valves. Depending on the actual sump design, flooding may eventually occur.

In summary, the consequences of stuck open SDV vent and drain valves are not as severe as those for a break. Timely operator action before the flooding reaches vital equipment levels will ensure the operability of equipment for detection and mitigation of the valves' failure.

However, since flooding from such an event is conceivable the probability of stuck open valves will be addressed. A typical configuration where the vent, drain and relief valves (if any) are piped to sumps, will be analyzed.

2.2.3.1 Failure Rate of Drain and Vent Valves

Both the drain and vent valves are air actuated globe valves which close upon loss of air. The air is controlled by solenoid operated valves. The vent and drain valves could remain open while the reactor is scrammed if (1) they stick open, (2) the air in them cannot vent , or (3) air from the instrument line is not cut off.

The probability of an air operated valve sticking open is 6.6×10^{-4}/demand (Ref. 8). The probability, then , of two drain or vent valves in series sticking open is 4.4×10^{-7} per demand. For the average of 9 scrams per year the probability is 3.9×10^{-6} per reactor year; for the maximum of 17 it is 7.4×10^{-6}/ry.

The air to the vent and drain valves are normally controlled by two solenoid operated valves configured as shown in figure 2.4. Solenoid valves V3 and V4 each controls one vent and one drain valve. Under normal operating conditions the exhaust port is closed and the other two ports are open. This maintains air pressure on the vent and drain valves to keep them open. When a scram occurs, the air supply port should close and the exhaust port open. This would allow the air from the drain and vent valves to escape and thus close. A failure, however, can be postulated where both the air supply and exhaust ports are plugged. This would prevent the air from the drain and vent valves from escaping and keep them in the open position.

The median probability of a solenoid valve being plugged is 8×10^{-5}/demand (Ref. 3). In order for two drain or two vent valves to fail open (1) both solenoid valves need to be plugged or (2) one solenoid valve must plug and one drain or vent valve, not controlled by the plugged solenoid valve, must stick open. The sum of the probabilities for the various combinations is 2.2×10^{-7}/demand. For 9 scrams/year it becomes 2×10^{-6}/ry, for 17 scrams/year it is 3.7×10^{-6}/ry.

Given a scram signal, the air to two drain or two vent valves is maintained only if all four valves fail in the no-scram position. The median probability for a solenoid valve to fail to operate is 1×10^{-3}/demand (Ref. 3). The probability for four valves to not operate is thus 1×10^{-12}/demand. Given 9 (17) scrams per year, the probability of the air not being cut off is 1×10^{-11} (2×10^{-11}).

In summary, then, the probability of either two drain or two vent valves failing open is 6×10^{-6}/ry for nine scrams a year and 1×10^{-5}/ry for 17 scrams a year.

2.2.3.2 Failure Rate of SDV Relief Valve

Some plants are equipped with an SDV relief valve as shown in figure 2.2. It was originally installed to comply with ANSI B31.1 for occasional over pressurizations. It was not, and is not specifically required for this system because the SDV pressure is limited to that of the reactor, which has its own pressure relief valves. The typical nominal opening set point is 1250 psig with a discharge capacity of 75 $\pm$ 25 gpm at 1375 psig. This flow rate is within the capability of most (if not all) sump pumps. For the valve to fail open, the pressure would have to exceed its setpoint and then it would have to fail to reseat. Events that will cause the pressure to exceed 1250 psig are transients such as closure of all Main Steam Isolation Valves (MSIV) with flux scram (i.e., failure of four scram position switches), or failure of several relief valves during a pressurization transient such as turbine trip without bypass.

To estimate the probability of stuck open SDV relief valve, consider the closure of all MSIV transient. The frequency of all MSIV closure with position switch scram is ~ 0.5/year (Ref. 9). The probability of a position switch failing is estimated to be 10^{-2}/demand (Ref. 3). Scram will not occur if two switches fail simultaneously; the probability is 10^{-4}/demand or 5×10^{-5}/year. The probability that a relief valve won't reseat is ~5×10^{-3}/ demand (Ref.8) (It is assumed to be similar to that for a primary relief valve) or ~ 2×10^{-3}/year. Thus the probability that the relief valve will stick open is ~1×10^{-7}/year for closure of all MSIV with flux scram.

The probability of a stuck open SDV relief valve for other events such as Turbine Trip without bypass with failure of several primary relief valves to open is even lower. The probability of the SDV sticking open is thus conservatively estimated to be 1×10^{-7}/year.

2.2.4 Other Considerations

Figure 2.2 shows that the SDV system has several calibration valves that are normally locked closed. In addition, the end of each calibration line is capped. The only credible way that a severe leak could occur from this line is either from a full break or from failure to fully close the valve and recap the line. The former event has already been included under pipe break. The latter depends on the quality of inservice inspection. The NRC through NUREG-0803 has mandated that ''surveillance, maintainance, inspection or modification procedures which conceivably have the potential for defeating SDV integrity be reviewed (or modified, if necessary) by licensee on a plant-by-plant basis. These plant-specific reviews should verify that all such procedures contain sufficient guidance to ensure that the loss of SDV system integrity will not occur at times when such integrity should be available.'' These actions should preclude the valve being left open and the end of the pipe being uncapped.

2.2.5 Probability of Breach of SDV integrity

The probability of loss of SDV integrity is the sum of the probabilities of pipe failure and valve failure. Based on the calculations previously discussed these probabilities are:

Failure mode	Probability/Reactor year Average Plant	Limiting Plant	
Pipe Break	1×10^{-7}	6×10^{-7}	(Table 2.2)
Vent valve open	6×10^{-6}	1×10^{-5}	(Section 2.2.3.1)
Drain valve open	6×10^{-6}	1×10^{-5}	(Section 2.2.3.1)
Relief valve open	1×10^{-7}	1×10^{-7}	(Section 2.2.3.2)
All other	Negligible	Negligible	(Section 2.2.3.3)
Total	$\sim 1.2 \times 10^{-5}$	$\sim 2 \times 10^{-5}$	

These values are based on the scrams not being reset.

NUREG-0803 conservatively estimated the probability of failure to reset scram in 30 minutes at ~.5. This high value was used because of the uncertainty in the post-leak environment that might contribute to the inability to reset.

This argument, however, is not as applicable in the case of stuck open vent or drain valves as it is to pipe break, since valves are not spraying uncontrollably in the air. Rather, they are discharging into sumps. In this case the operator failure to reset will most likely be the dominant failure-to-reset.

NUREG-0803 used an upper bound value of 0.02 for operator failure to reset.

Thus, using a failure to reset probability of 0.5 in the case of pipe breaks and 0.02 in the case of valve failures, the probabilities of non-isolatable leaks are:

Failure Mode	Probability/Reactor year Average Plant	Limiting Plant
Pipe Break	5×10^{-8}	3×10^{-7}
Vent Valve Open	1.2×10^{-7}	2×10^{-7}
Drain Valve Open	1.2×10^{-7}	2×10^{-7}
Relief Valve Open	2×10^{-9}	2×10^{-9}
Total	$\sim 3.0 \times 10^{-7}$	$\sim 7 \times 10^{-7}$

3.0 Summary and Conclusions

NUREG-0803 requires the equipment used to detect and/or mitigate the consequences of a loss of SDV integrity event be qualified for the environmental conditions of that event. This study concludes that environmental qualification is not necessary due to the low probability of a breach in SDV integrity. It also follows that there is a low probability of core damage resulting from such a breach.

The loss of SDV integrity can occur from any of four failure modes: (1) rupture of the SDV piping upstream of the vent and drain valves, (2) failure of the redundant vent valves to close following a scram, (3) failure of the redundant drain valves to close following a scram or (4) failure of the SDV relief valve. The first failure mode was investigated using methods similar to those used in NUREG-0803 and NEDO-24342. Actual plant data on SDV pipe size and scram frequency was considered for these two approaches. The calculated break probabilities from these two approaches was compared to the calculated probability using a fracture mechanics approach and the results were shown to be consistent.

The probabilities associated with failure of the vent or drain valves to close were calculated based on previous operating history with this type of valve. The probability of an SDV relief valve failure to close was small relative to the other failure modes due to the relatively low frequency of challenge to this valve.

Consideration was given in the probability analysis to the ability of the operator to reset the scram. Due to the more severe environmental conditions, that probability is lower for the SDV pipe break than for the vent or drain valve failure.

The total probability of a breach in SDV integrity is the sum of the individual probabilities for each failure mode. That total probability was determined to be approximately 3×10^{-7} per reactor year.

The probability of a core melt event given the breach in SDV integrity was previously calculated and reported in Section 7.8 of NEDO 24342 and was determined to be 1.2×10^{-4} per plant year. Therefore, the probability of a breach in SDV integrity leading to a core melt is approximately 4×10^{-11} per plant year. This is significantly below the NRC proposed safety goal for core melt events which is 10^{-4} per reactor year.

The NRC, in NUREG-0803, stated that ''it was agreed that if the probability of core damage from the postulated scenario (i.e., loss of SDV pipe integrity) was shown to be sufficiently small, no further review, beyond verification of plant-specific response applicability, would be necessary''. They further noted that ''as the review progressed, it became evident that a sufficient data base did not exist to conservatively terminate the generic review on the basis of a quantitative risk assessment''. However, considering that the estimated core melt frequency following a loss of SDV integrity is considerably below the proposed NRC safety goal (by ~6 orders of magnitude), this significant margin should be sufficient to account for any perceived sparsity in the data base.

Therefore, it is concluded that the breach of SDV integrity need not be considered for environmental qualification of equipment in the reactor building.

Table 2.1 - Characteristics of the [illegible]V System for the Various Plants

Parameter	Fermi 2	PB 2	PB 3	Duane Arnold	Lime-rick	Fitz	Pil-grim	WNP 2	Hatch 2	Oyster Creek	Susque-hanna	Monti-cello	NMP 1	Bruns-wick 1 + 2
Length of Pipe(ft)														
1/2 -<2 ''	1700	2023	2053	997	1439	1037	1015	1670	1684	1548	1992	1108	949	1761
2''-6''	120	582	9	158	140	18	370	293	123	278	181	244	327	303
> 6''	290	11	414	188	170	257	18	147	274	100	289	71	94	241
Instal. Exam. Class	2	1	1	2	2	B31.1	B31.1	1	-	B31.1	(5)	2	-	2
Desg Code + Class	2	B31.1 + GE	B31.1 + GE	1	2	B31.1 + GE	2	Safety 2 Qual. 1	2	(3)	2	B31.1 + GE	B31.1 + Class 1	B31.1 + GE
Seismic Design Class	1	1	1	1	1	1	2	1	1	(4)	1	1	1	1
In Serv. Insp. Class	2(1)	1	1	1	2	2	ASME XI	ASME XI	2	Surveill for water	2	1	1	None
Welded Joints	1044	-	-	941	~905	1044	974	1205	683	957	1097	833	-	1024
Threaded Joints	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Average Scram/yr	(2)	4.3	7.5	8.2	(2)	7.3	9.5	(2)	17	-	(2)	6.8	12.6	17*
Average Scram Dur. min.	(2)	17.5	17.5	5.83	(2)	-	30	(2)	-	-	(2)	16	1	4

( ) - Number in paranthesis refers to Note.
- - Not Available
* - Average scram/yr for both Brunswick 1 and 2

Notes For Table 2.1

1) Visual test all piping while at hydrostatic pressure. Ultrasonic test scram discharge volume and instrument volume (25% of stress welds over 10 years). Frequency is refueling cycle and Class 2 program.

2) Plant has not started up yet, so there is no scram data.

3) ASA B31.1, ASME I and VIII and ASME Sections III and XI.

4) Uniform Building Code with following acceleration values:

   .43g Horiz. .29g Vert.

5) VT/PT for withdrawl lines, VT/RT for headers and instrument volume.

TABLE 2.2 - BREAK PROBABILITIES USING EXPERIENCE APPROACH

Parameter	Average Plant	Limiting Plant
Length of SDV pipe (ft)		
1/2'' to 2'' diam.	1496	2023
2'' to 6'' diam.	225	582
>6'' diam.	183	11
Scrams/year	9	17
Total time to reset per year (min)	91	285
Fraction of time scrammed(1)	1.7×10^{-4}	5.4×10^{-4}
Probability (NEDO-24342)(2)	1.3×10^{-7}/reactor year	6×10^{-7}/reactor year
Probability (NUREG-0803)(3)	1.0×10^{-7}/reactor year	4.2×10^{-7}/reactor year

( ) - refers to Notes.

Notes For Table 2.2

1) Fraction of time scrammed is the Total time to reset per year divided by the number of minutes in a year.

2) Probability (NEDO-24342)

$$=[(L_1 \times 10^{-3}) + (L_2 \times 3\times10^{-4}) + (L_3 \times 10^{-4})] \times F_1/2000$$

where: L_1 = Length of SDV piping of 1/2'' to 2'' diameter

L_2 = Length of SDV piping of 2'' to 6'' diameter

L_3 = Length of SDV piping of >6'' diameter

F_1 = Fraction of time scrammed

3) Probability (NUREG-0803)

$$= (L_1 + L_2 + L_3) \times 3 \times10^{-7} \times F_1$$

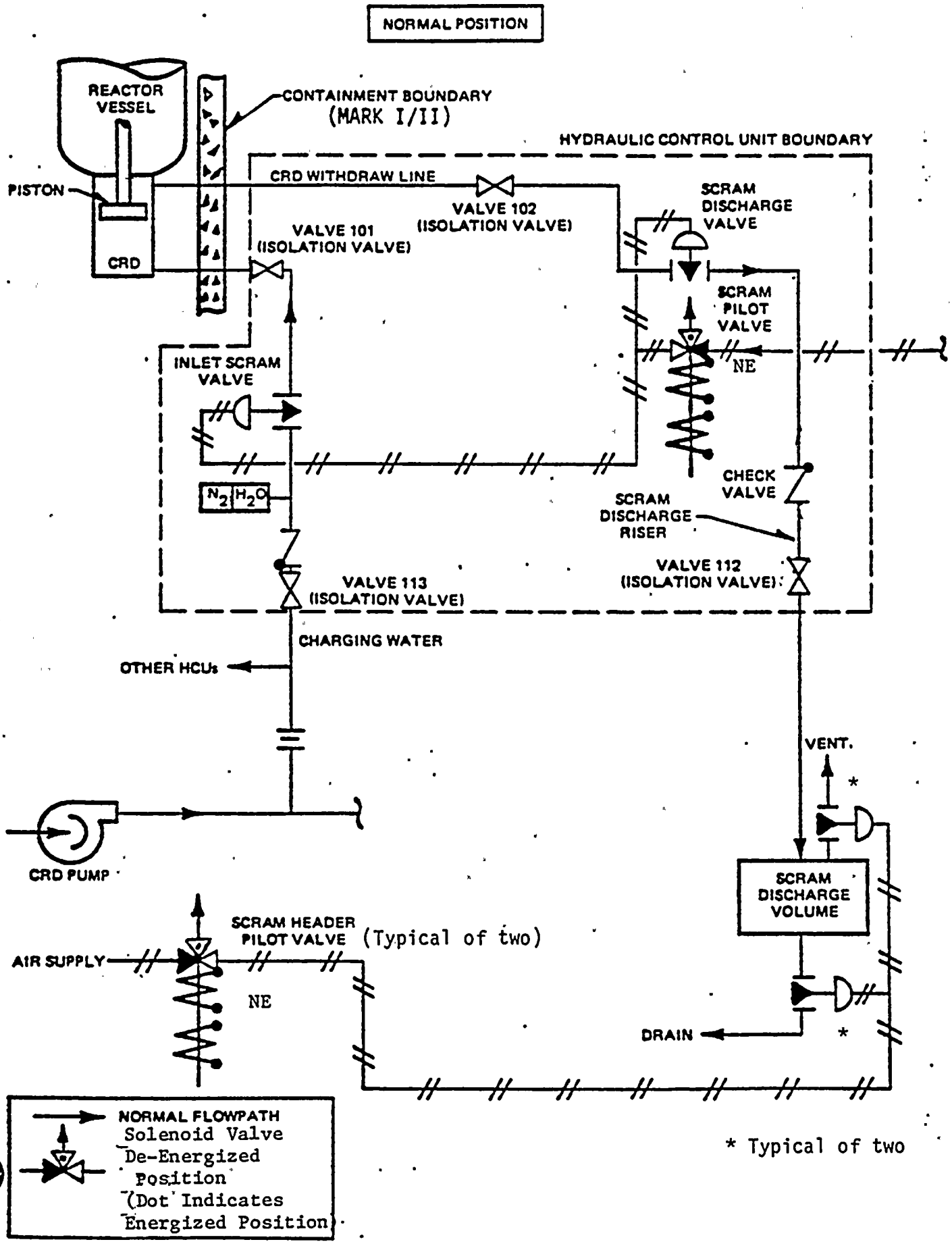

Figure 2.1 Simplified Schematic of Control Rod Drive System

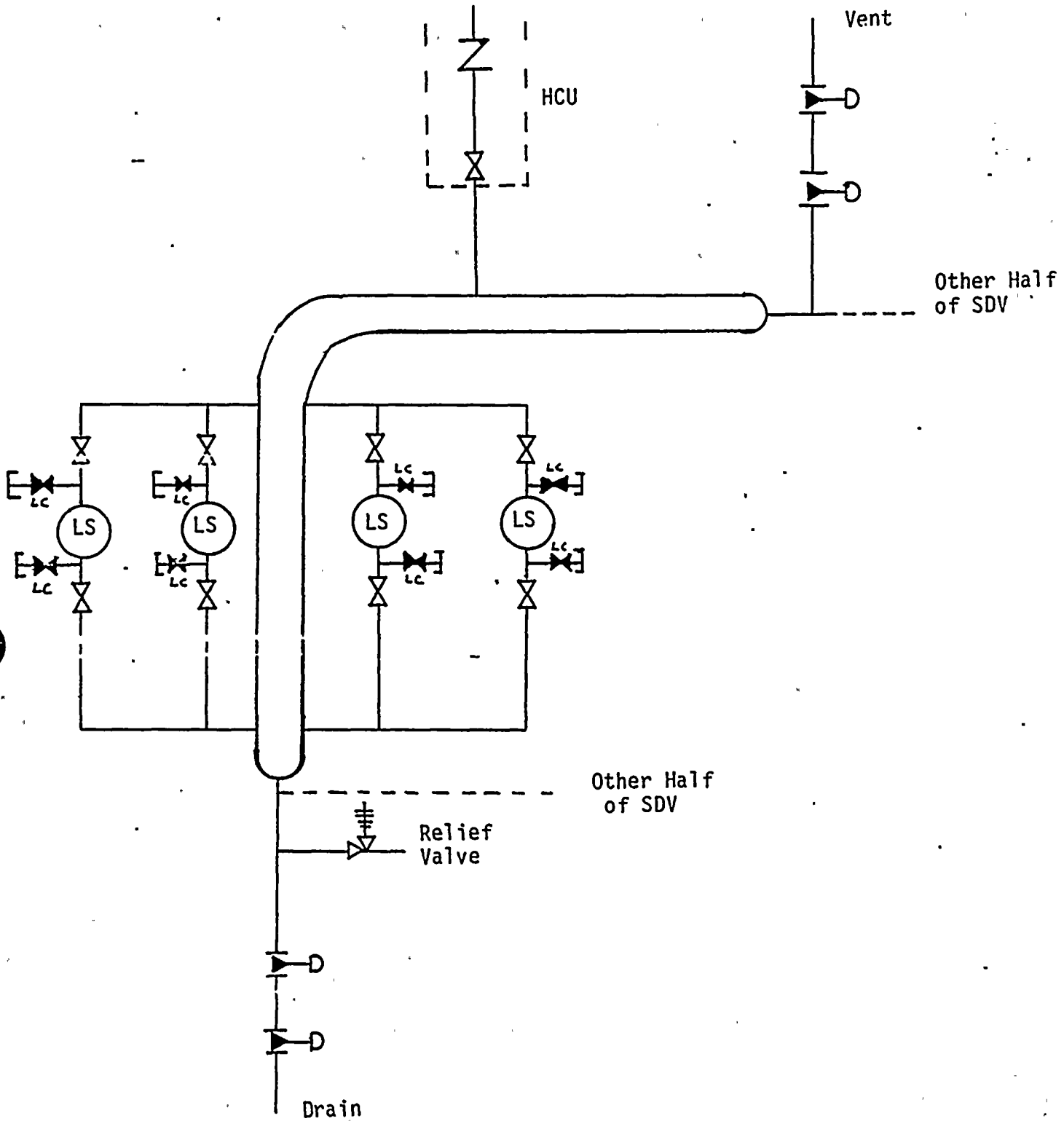

Figure 2.2 - Typical Scram Discharge Volume Configuration (Simplified)

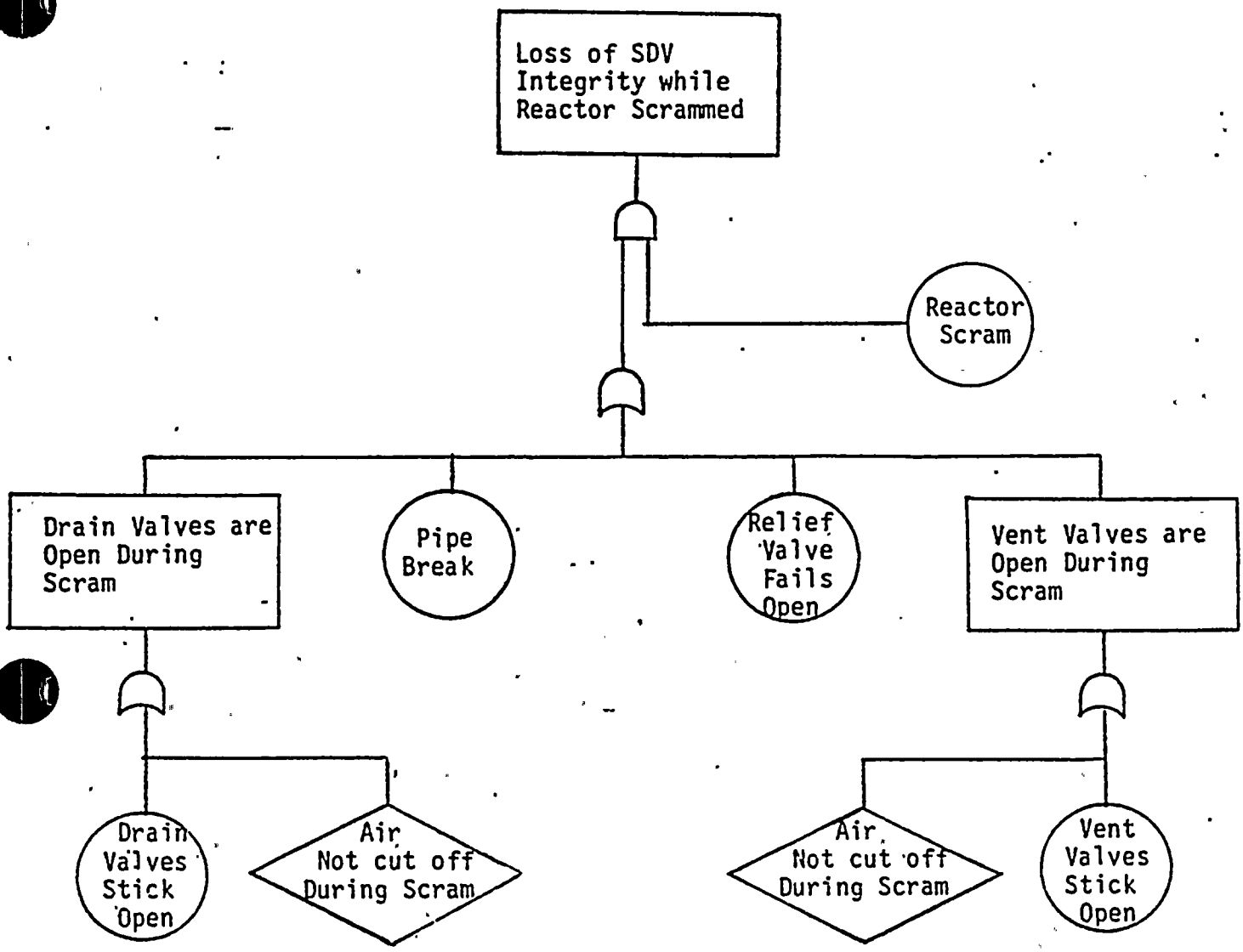

Figure 2.3 - Fault Tree For Loss of SDV Integrity

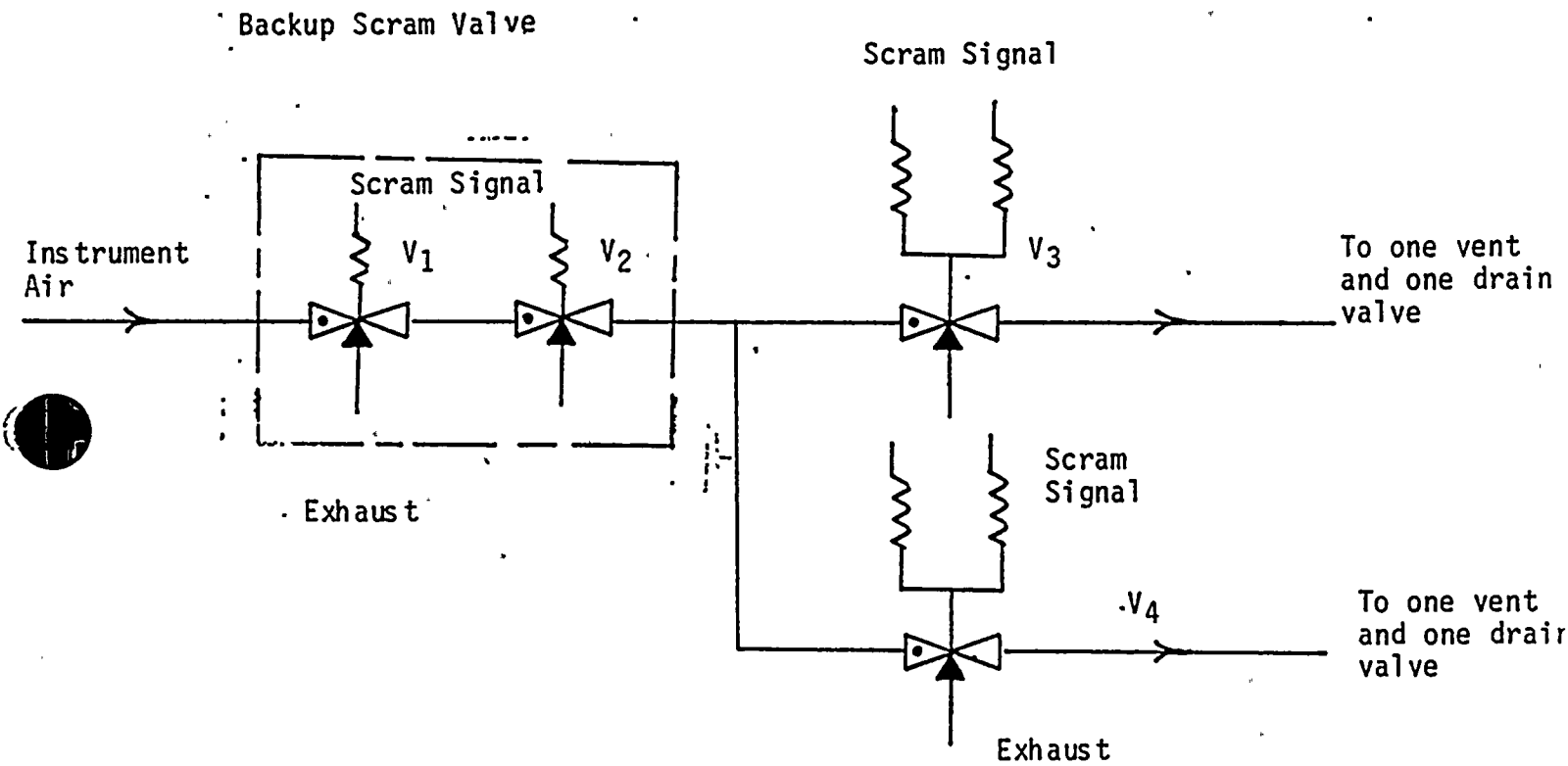

Figure 2.4 Simplfied Diagram Of A Typical SDV Instrument Air Control. The position shown is the no-scram position. The dot represents the port that will close upon receipt of the scram signal.

4.0 References

1) 'Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping', NUREG-0803, August 1981.

2) L.F. Fidrych, R.L. Gridley, 'GE Evaluation In Response To NRC Request Regarding BWR Scram System Pipe Break', NEDO-24342, April, 1981

3) 'Reactor Safety Study', WASH-1400, (NUREG-75/014) October 1975.

4) NRC Memorandum, 'Generic Safety Evaluation Report - BWR Scram Discharge System', December 1980.

5) Farmer, F.G. et.al., 'Screening Values For National Reliability Evaluation Program Reliability Applications', Preliminary Draft, April 1982.

6) 'Review and Assessment of Research Relevant to Design Aspects of Nuclear Power Plants Piping Systems', NUREG-0307, July, 1977.

7) J.S. Abel, 'Quad Cities Station Units 1 and 2, Dresden Station Units 2 and 3 Plant Specific Response to NUREG-0803', letter to T.J. Rausch, Jan. 25, 1982.

8) 'Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants', NUREG/CR-1363 vol.3.

9) 'ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients,' EPRI NP-2230, January 1982.

10) 'Safety Goals For Nuclear Power Plants: A Discussion Paper', NUREG-0880, February 1982 (Draft).

APPENDIX A

Participating Utilities	Plant
Boston Edison Co.	Pilgrim
Carolina Power + Light Co.	Brunswick 1 and 2
Detroit Edison Co.	Fermi 2
Georgia Power Co.	Hatch 2
GPU Nuclear	Oyster Creek
Iowa Electric Light and Power Co.	Duane Arnold
Niagara Mohawk Power Co.	Nine Mile Point 1
Northeast Utilities	Millstone
Northern States Power Co.	Monticello
PASNY	Fitzpatrick
Pennsylvania Power + Light Co.	Susquehanna 1 and 2
Philadelphia Electric Co.	Peach Bottom 2 Peach Bottom 3 Limerick 1 and 2
Public Service Electric + Gas Co.	Hope Creek 1
Washington Public Power Supply System	WNP-2

APPENDIX C

SDV BREAK MITIGATION COMPONENTS

REQUIRING EQUIPMENT QUALIFICATION

TABLE C-I

Instruments Required to Mitigate Effects of SDV Leak

Device	System
FS-E11-1N021	Residual Heat Removal (RHR)
FT-E11-1N007	RHR
FT-E11-1N007	RHR
FT-E11-1N015	RHR
PS-E11-1N016	RHR
PS-E11-1N020	RHR
PS-E11-1N020	RHR
PT-E21-1N001	Core Spray (CS)
FT-E21-1N003	CS
FIS-E21-1N006	CS
PS-E21-1N008	CS
PS-E21-1N009	CS
FIS-B-21-1N006	Nuclear Boiler (NB)
FIS-B-21-1N007	NB
FIS-B-21-1N008	NB
FIS-B-21-1N009	NB
TE-B21-1N010	NB
TE-B21-1N014	NB
TE-B21-1N016	NB
PS-B21-1N015	NB
PS-B21-1N021B	NB
PS-B21-1N021D	NB
PS-B21-1N021A, E	NB

Device	System
PS-B21-1N021C, G	NB
PS-B21-1N022	Nuclear Boiler
PS-B21-1N023	NB
LIS-B21-1N025	NB
LIS-B21-1N026	NB
LT-B21-1N027	NB
PVT-B21-1N032	NB
PT-B21-1N055	NB
PS-B21-1N045	NB

TABLE C-II

EQUIPMENT USED TO MITIGATE THE EFFECTS OF AN

SDV LEAK LOCATED IN THE BREAK ENVIRONMENT

- RHR Discharge to Suppression Pool -

Equipment	Description	Power Supply
"B" Train		
HV-E11-1F004B	RHR Pump B Suction Valve	1B226
HV-E11-1F004D	RHR Pump D Suction Valve	1B247
HV-11210B	RHR HX Tube Side Inlet Valve	1B247
HV-11215B	RHR HX Tube Side Outlet Valve	1B247
HV-1F047B	RHR HX Shell Side Inlet Valve	1B226
HV-1F048B	RHR HX Shell Side Bypass Valve	1B247
HV-1F028B	RHR Discharge To Supp. Pool ISO	1B226
	Valve	1B226
HV-1F024B	RHR Discharge to Supp. Pool ISO Valve	1B247
HV-1F003B	RHR HX Outlet Valve	1B226
1V210B	RHR B Pump Unit Cooler	1B226
1V210D	RHR D Pump Unit Cooler	1B247
1P202B	RHR B Pump	1A202
1P202D	RHR D Pump	1A204
"A" Train		
HV-E11-1F004A	RHR Pump A Suction Valve	1B216
HV-E11-1F004C	RHR Pump C Suction Valve	1B237
HV-11210A	RHR HX Tube Side Inlet Valve	1B237

Equipment	Description	Power Supply
HV-11215A	RHR HX Tube Side Outlet Valve	1B237
HV-E11-1F024A	RHR Discharge to Supp. Pool ISO Valve	1B237
HV-E11-1F047A	RHR HX Shell Side Inlet Valve	1B216
HV-E11-1F048A	RHR HX Shell Side Bypass Valve	1B237
HV-E11-1F028A	RHR Discharge to Supp. Pool ISO Valve	1B216
HV-E11-1F003A	RHR HX Outlet Valve	1B216
1V210A	RHR A Pump Unit Cooler	1B216
1V210C	RHR D Pump Unit Cooler	1B237
1P202A	RHR A Pump	1A201
1P202C	RHR C Pump	1A203
- RHR LPCI MODE -		
"B" Train		
HV-E11-1F004B	RHR Pump B Suction Valve	1B226
HV-E11-1F004D	RHR Pump D Suction Valve	1B247
HV-E11-1F017B	RHR Inject. Outbd. ISO Valve	1B226
HV-E11-1F015B	RHR Inject. Outbd. ISO Valve	1B229
HV-E11-1F031B	RRP B Discharge Valve	1B229
1V210B	RHR B Pump Unit Cooler	1B226
1V210D	RHR D Pump Unit Cooler	1B247
1P202B	RHR B Pump	1A202
1P202D	RHR D Pump	1A204
"A" Train		
HV-E11-1F004A	RHR Pump A Suction Valve	1B216
HV-E11-1F004C	RHR Pump C Suction Valve	1B237
HV-E11-1F017A	RHR Inject. Outbd. ISO Valve	1B216

Equipment	Description	Power Supply
HV-E11-1F015A	RHR Inject. Outbd. ISO Valve	1B219
HV-E11-1F031A	RRP A Discharge Valve	1B219
1V210A	RHR A Pump Unit Cooler	1B216
1V210C	RHR C Pump Unit Cooler	1B237
1P202A	RHR A Pump	1A201
1P202C	RHR C Pump	1A203
- CORE SPRAY SYSTEM -		
"B" Train		
HV-E21-1F001B	CS Supp. Pool Suction Valve	1B226
HV-E21-1F004B	CS Outbd. ISO Valve	1B227
HV-E21-1F005B	CS Outbd. ISO Valve	1B227
1V211B	CS B Pump Unit Cooler	1B226
1V211D	CS D Pump Unit Cooler	1B247
1P206B	CS B Pump	1A202
1P206D	CS D Pump	1A204
"A" Train		
HV-E21-1F001A	CS Supp. Pool Suction Valve	1B216
HV-E21-1F004A	CS Outbd. ISO Valve	1B217
HV-E21-1F005A	CS Outbd. ISO Valve	1B217
1V211A	CSA Pump Unit Cooler	1B216
1V211C	CS C Pump Unit Cooler	1B237
1P206A	CS A Pump	1A201
1P206C	CS C Pump	1A203

TABLE C-III

EQUIPMENT USED TO MITIGATE THE EFFECTS OF AN SDV BREAK

LOCATED OUTSIDE THE BREAK EQUIPMENT*

1) AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) - All components associated with the function of this system are located inside primary containment or outside the reactor building, with the exception of cabling which is not susceptible to the break environment. Therefore, this system will not be effected by the environment and its environmental qualification does not need to be considered.

2) Condensate System - All components associated with the function of this system are located in the turbine building which is outside of the break environment. Therefore, the environmental qualification of this equipment does not need to be considered.

3. Turbine Building Closed Cooling Water - All components of this system are located within the turbine building, which is outside the break environment. Therefore, the environment qualification of this equipment does not need to be qualified.

4. RHR Service Water System: All Components of this system, except for the Heat Exchanger Tube side inlet and outlet valves indicated in Table C-II, are located outside the reactor building and are not subject to the SDV break environment.

* Note: Equipment located outside of the SDV break does not require equipment qualification.

5. Emergency Service Water System: All components of this system are located outside the Reactor Building and therefore are not subject to the SDV break environment.

APPENDIX D

EQUIPMENT QUALIFICATION OF MITIGATION/IDENTIFICATION COMPONENTS

APPENDIX D

Equipment Qualification

Safety-related equipment used to mitigate the effects of an SDV leak are indicated below. Maximum SDV leak environment temperatures and qualified environments for equipment are given. Radiation and pressure increases due to SDV leakage are negligible.

1. Instrumentation - Instruments and their environmental qualification parameters are given in Table I.

2. Valve Operators - Limitorque valve motor operators outside containment are qualified to 250°F for greater than 4 hours. Peak SDV break temperature is 198°F.

3. Pump Motors - General Electric core spray and residual heat removal pump motors are qualified to 130°F ambient temperature for 100 days operation. SDV break temperature is 110°F.

4. Unit Cooler Motors - Westinghouse unit cooler motors have class H insulation rated at 180°C. Motorettes were successfully tested at 210°C for over 7,500 hours. SDV break temperature is 110°F and 198°F.

5. Switchgear - Westinghouse 4.16KV switchgear is rated at 104°F for continuous operation. SDV break temperature is 210°F.

6. 480 Volt Motor Control Centers - Cutler Hammer Motor Control Centers have been qualified by analysis to 120°F for 100 days. Rated temperature is 104°F. Ambient temperatures at MCC's due to SDV leakage would be 95°F, 102°F, 110°F, 120°F, 135°F and 198°F.

   MCC starter components which are susceptible to the effects of increased temperatures are molded case circuit breakers, control circuit fuses, auxiliary relays, contactors, and thermal overload relays.

   Circuit breakers used for MOV's and unit coolers are magnetic only types and their operation is not significantly affected by increased temperatures. Operation of thermal overload relays is affected by increased ambient temperatures. On valve motor operators thermal overload relays are bypassed during plant operation; they are used only during testing. Thermal overload relays for unit cooler starters may trip on increased ambient temperatures.

APPENDIX D

Table I

Instruments Required to Mitigate Effects of SDV Leak

Device	System	Manu	Mod	E.Q. Temp	Max 0803 Temp	EQEL	Qual. Temp.
FS-E11-1N021	RHR	ITT Barton	288,289	130°F	125°F	26	212°F/6 hrs.
FT-E11-1N007	RHR	Rosemount	1151DP	130°F	120°F	48	350°F
FT-E11-1N015	RHR	Rosemount	1151DP	130°F	120°F	48	350°F
PS-E11-1N016	RHR	Static O Ring	5N,6N	130°F	115°F	10	130°F
PS-E11-1N020	RHR	Static O Ring	5N,6N	130°F	115°F	10	130°F
PT-E21-1N001	CS	Rosemount	1151DP	130°F	125°F	48	350°F
FT-E21-1N003	CS	Rosemount	1151DP	130°F	125°F	48	350°F
FIS-E21-1N006	CS	Barton	288,289	104°F	125°F	26	212°F/6 hrs.
PS-E21-1N008	CS	Barksdale	PIH	130°F	110°F	31	212°F/6 hrs.
PS-E21-1N009	CS	Barksdale	PIH	130°F	110°F	31	212°F/6 hrs.
FIS-B-21-1N006	NB	Barton	288	104°F	137°F	26	212°F/6 hrs.
FIS-B-21-1N007	NB	Barton	288	104°F	137°F	26	212°F/6 hrs.
FIS-B-21-1N008	NB	Barton	288	104°F	137°F	26	212°F/6 hrs.
FIS-B-21-1N009	NB	Barton	288	104°F	137°F	26	212°F/6 hrs.
TE-B21-1N010	NB	Pyco	-	130°F	125°F	19	130°F
TE-B21-1N014	NB	Pyco	-	130°F	125°F	19	130°F
TE-B21-1N016	NB	Pyco	-	130°F	125°F	19	130°F
PS-B21-1N015	NB	Barksdale	BIT	130°F	125°F	54	130°F
PS-B21-1N021B	NB	Barton	288	104°F	137°F	26	212°F/6 hrs.
PS-B21-1N021D	NB	Barton	288	104°F	108°F	26	212°F/6 hrs.
PS-B21-1N021A, E,	NB	Barksdale	BIT	104°F	137°F	54	212°F/6 hrs.
PS-B21-1N021C, G	NB	Barksdale	BIT	104°F	137°F	54	212°F/6 hrs.

APPEND
Table
Instruments Required to Mitigate Effects of SDV Leak

Device	System	Manu	Mod	E.Q. Temp	Max 0803 Temp	EQEL	Qual. Temp.
PS-B21-1N022	NB	Barksdale	BIT	104°F	137°F	54	212°F/6 hrs.
PS-B21-1N023	NB	Barksdale	BIT	104°F	137°F	54	212°F/6 hrs.
LIS-B21-1N025	NB	Yarway	4418C	104°F	137°F	27	250°F
LIS-B21-1N026	NB	Barton	760	104°F	137°F	25	212°F/6 hrs.
LT-B21-1N027	NB	Rosemount	1151DP	104°F	137°F	48	350°F
PVT-B21-1N032	NB	Rosemount	1151DP	104°F	137°F	48	350°F
PT-B21-1N055	NB	Rosemount	1151DP	104°F	137°F	48	350°F
PS-B21-1N045	NB	Barksdale	BIT	104°F	137°F	54	212°F/6 hrs.

APPENDIX D

Table II

480 VAC Power Supplies for SDV Leak Mitigation Equipment with Loads

MCC 1B2/6 @ 120°F

HV-E11-1F004A	RHR Pump A Suction Valve
HV-E11-1F047A	RHR HX Shell Side Inlet Valve
HV-E11-1F028A	RHR Discharge to Supp. Pool ISO Valve
HV-E11-1F003A	RHR HX Outlet Valve
HV-E21-1F001A	CS Supp Pool Suction Valve
1V210A	RHR A Pump Unit Cooler
HV-E11-1F017A	RHR Inject Outboard ISO Valve
1V211A	CS A Pump Unit Cooler

MCC 1B217 @ 104°F

HV-E21-1F004A	CS Outb'd ISO Valve
HV-E21-1F005A	CS Outb'd ISO Valve

MCC 1B219 @ 135°F

HV-E11-1F015A	RHR Injection Outb'd

	ISO Valve
HV-B31-1F031A	RRP A Discharge Vlv.

MCC 1B226 @ 102°F

HV-E11-1F004B	RHR Pump Suction Vlv.
HV-1F047B	RHR HX Shell Side Inlet Valve
HV-1F028B	RHR Discharge to Supp. Pool ISO Valve
HV-1F003B	RHR HX Outlet Vlv.
1V210B	RHR Pump Room Unit Cooler
HV-E21-1F001B	CS Suppression Pool Suction Valve
1V211B	CS Valve B Pump Unit Cooler
HV-E11-1F017B	RHR Inject Outb'd ISO Valve

MCC 1B227 @ 104°F

HV-E21-1F004B	CS Outb'd ISO Valve
HV-E21-1F005B	CS Outb'd ISO Valve

MCC 18229 @ 198°F

HV-E11-1F015B	RHR Inject Outb'd ISO Valve
HV-B31-1F031B	RRP B Discharge Vlv.

MCC 1B 237 @ 135°F

HV-E11-1F004C	RHR Pump C Suction Valve
HV-11210A	RHR HX Tube Side Inlet Valve

HV-11215A	RHR HX Tube Side Outlet Valve
HV-E11-1F024A	RHR Discharge to Supp. Pool ISO Valve
HV-E11-1F048A	RHR HX Shell Side Bypass Valve
1V210C	RHR C Pump Unit Cooler
1V211C	CS C Pump Unit Cooler

MCC 1B247 @ 110°F

HV-E11-1F004D	RHR Pump D Suction Valve
HV-11210B	RHR HX Tube Side Inlet Valve
HV-11215B	RHR HX Tube Side Outlet Valve
HV-1F048B	RHR HX Shell Side Bypass Valve
HV-1F024B	RHR Discharge to Supp. Pool ISO Valve
1V210D	RHR D Pump Unit Cooler
1V211D	CS D Pump Unit Cooler

APPENDIX E

SINGLE COLD SHUTDOWN AND

LEAK DETECTION PATH

(1) Automatic Depressurization System

(2) Condensate System

- Condensate pumps
- Valves
- Main Condenser

(3) Circulating Water Systems

- Circulating water pumps and associated valves

(4) Turbine Building Closed Cooling Water System

(5) Service Water System

(6) Power is supplied from the Startup transformer

(7) CRD Temperature Indicator

(8) CRD ROD POSITION Indication System

APPENDIX F

JUSTIFICATION FOR INTERIM OPERATOR FORMS

ENVIRONMENTAL QUALIFICATION - JUSTIFICATION FOR INTERIM OPERATION

Component Ident. No(s).: MCC-1B219

Component Name: _____

System: _____ Purchase Order: _____

(1) Component(s) Safety Function:

This MCC supplies power to the following safety related notor operated valves:

HV-E11-1F015A-RHR Injection Outboard Isolation Valve

(2) Justification for Interim Operation:

This MCC does not supply equipment which is on the single cold shutdown path. Therefore, failure of this MCC will not effect the ability to safety shutdown the plant.

(3) Interim Operation is X Justified ____ Not Justified

ENVIRONMENTAL QUALIFICATION - JUSTIFICATION FOR INTERIM OPERATION

Component Ident. No(s).: MCC-1B229

Component Name: _____

System: _____ Purchase Order: _____

(1) Component(s) Safety Function:

This MCC supplies power for the following safety related motor operated valves:

HV-E11-1F015B - RHR Injection Outboard Isolation Valve
HV-E11-1F031B - RRP "B" Discharge Valve

(2) Justification for Interim Operation:

This MCC does not supply equipment which is on the single cold shutdown path. Therefore failure of this MCC will not effect the ability to safely shutdown the plant.

(3) Interim Operation is X Justified ____ Not Justified

ENVIRONMENTAL QUALIFICATION - JUSTIFICATION FOR INTERIM OPERATION

Component Ident. No(s).: MCC-1B237

Component Name: _____

System: _____ Purchase Order: _____

(1) Component(s) Safety Function:

This MCC supplies power to the following safety related motor operated valves:

HV-E11-1F004C - RHR PUMP C SUCTION VALVE
1V210C - RHR "C" PUMP UNIT COOLER
HV-11210A - RHR HX TUBE SIDE INLET VALVE
HV-11215A - RHR HX TUBE SIDE OUTLET VALVE
HV-E11-1F024A - RHR DISCHARGE TO SUPPRESSION POOL ISOLATION VALVE
HV-E11-1F048A - RHR HX SHELL SIDE BYPASS VALVE
1V211C - CORE SPRAY "C" PUMP UNIT COOLER

(2) Justification for Interim Operation:

This MCC does not supply equipment which is on the single Cold shutdown path. Therefore, failure of this MCC will not effect ability to safely shutdown the plant.

(3) Interim Operation is X Justified ____ Not Justified

ENVIRONMENTAL QUALIFICATION - JUSTIFICATION FOR INTERIM OPERATION

Component Ident. No(s).: 1A201, 1A202, 1A203, 1A204

Component Name: 4.16kv Emergency Switchgear

System: _____ Purchase Order: _____

(1) Component(s) Safety Function:

These switchgear supply power to the following safety related pump motors:

1P202A-D-RHR Pump Motors

1P206A-D-Core Spray Pump Motors

(2) Justification for Interim Operation:

This switchgear does not supply equipment which is on the single cold shutdown path. Therefore, failure of this switchgear will not effect the ability to safely shutdown the plant.

(3) Interim Operation is X Justified ____ Not Justified