SHOREHAM EVALUATION REPORT

REGARDING INTEGRITY OF SCRAM

SYSTEM PIPING

EXECUTIVE SUMMARY

In August 1981, the U.S. Nuclear Regulatory Commission published NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping." This document addressed the possibility of scram system pipe breaks outside the primary containment. Specifically, a generic BWR probabilistic risk assessment in that document indicated that the postulated SDV event is not a dominant contributor to the probability of core damage. However, NRC guidance in Chapter 5 of NUREG-0803 requires that the assumptions used in the risk assessment be verified on a plant specific basis.

The plant specific issues in NUREG-0803 are addressed for Shoreham Nuclear Power Station (SNPS) in this document. It is established that:

1. The SDV piping satisfies all appropriate ASME Codes, is seismically qualified and a probabilistic fracture mechanics evaluation of the SDV system piping demonstrates that the probability of a pipe rupture in this system is not a significant contributor to an accident scenario;
2. For the unlikely event of a postulated break in the SDV system piping SNPS leak detection equipment and the associated operating procedures will guide the reactor operators to a prompt and successful mitigation of the event; and
3. The equipment needed to mitigate a postulated SDV pipe break event is tested for the environment and will function adequately to bring the reactor system to a safe shutdown condition.

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1. INTRODUCTION

1.1 BACKGROUND AND SUMMARY OF CONCERNS

In March 1981, the U.S. Nuclear Regulatory Commission (NRC) completed NUREG-0785, "Safety Concerns Associated with Pipe Breaks in BWR Scram Systems." This document addressed the possibility of scram system breaks located outside primary containment, and specifically within the Scram Discharge Volume (SDV) subsystem. It is acknowledged in NUREG-0785 that the resulting small leak flow lost from the bottom of the reactor vessel could easily be replaced by the High Pressure Coolant Injection (HPCI) system or by the Reactor Core Isolation Cooling (RCIC) system. Alternatively, the reactor vessel could be depressurized and inventory replaced by various low pressure systems. However, it was further postulated in NUREG-0785 that water emerging from the break could potentially run across the floor on which the SDV is located, flow down or through various stairwells, and eventually make its way to the basement, where essential ECCS pumps are located. Flooding of these pumps could conceivably impair the equipment itself and/or instrumentation systems required to assure long-term core cooling and water inventory replacement.

In April 1981, General Electric Company issued NEDO-24342, "GE Evaluation in Response to NRC Request Regarding BWR Scram," as a generic evaluation of these issues. This document concluded that SDV rupture does <u>not</u> constitute a significant safety problem because:

- Such a pipe break has an extremely low probability of occurrence.
- <u>Even if</u> such a break should occur, the event would be detected and terminated by manual operator action. Further, no lower level pump rooms would be flooded because manual depressurization would assure that any leak rate would be reduced to a level well within the capability of drain sump pumps.
- <u>Even if</u> all ECCS pumps should become unavailable, the long-term expected leak rate of 40-50 gpm could easily be replaced by other pumps.

In August 1981, the NRC released NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," which (1) evaluates the generic conclusions of NEDO-24342, (2) performs additional probabilistic analyses to show that the postulated event is not a dominant contributer to core damage, and (3) outlines certain plant-specific issues to be addressed by BWR owners. Continued plant operation is concluded to be acceptable because the risk level is lower than other events (e.g., Anticipated Transients Without Scram) which are under continuing study.

1.2 PLANT-SPECIFIC ISSUES IDENTIFIED BY NUREG-0803

The plant-specific resolution of the SDV matter raised by the NRC involves issues falling under three areas of concern: (1) Piping Integrity, (2) Mitigation Capability, and (3) Environmental Qualification.

1.2.1 Piping Integrity (PI)

The Quality Assurance (QA) for SDV piping and pipe supports must be checked by verifying that a seismic analysis has been performed and by verifying its as-built configuration. Inservice inspection procedures must be shown adequate for keeping the pipe failure probability at a low level.

1.2.2 Mitigation Capability (M)

Assuming that an SDV piping break could occur, the capability to mitigate the pipe break transient so that no radioactive release occurs in excess of the levels specified by 10 CFR 100 must be shown to exist. It must be verified that adequate operation procedures and instrumentation exist to ensure that such a condition is diagnosed and that appropriate mitigating action is taken to guarantee that adequate reactor pressure vessel (RPV) water inventory is maintained. Radiation levels in the reactor building should be low enough to permit operator access to the affected area, and offsite dose levels must at all times remain below limits set by 10 CFR 100. Finally, there must be adequate core cooling capacity, through ECCS and other pumps, to maintain water inventory at all times; this question is closely tied to equipment qualification issues discussed in Section 1.2.3.

1.2.3 Equipment Qualification (EQ)

On a plant-specific basis, the equipment must be identified which is necessary to detect a SDV pipe break or to mitigate the transient resulting from an unisolable SDV pipe break. The environmental conditions, including temperature, humidity, and wetting, must be assessed and the qualification status of all such equipment determined in relation to these environmental conditions as appropriate.

All of the above issues were addressed by GE in their NEDO-24342 document on a generic basis. The plant specific SDV issues are addressed for Long Island Lighting Company's (LILCO) Shoreham Nuclear Power Station (SNPS) in this report.

In addition to addressing the issues raised by the NRC a fracture mechanics analysis of the Shoreham SDV and the associated piping system shows that the probability of a SDV failure in this plant is sufficiently small that the concerns expressed in NUREG-0785 and NUREG-0803 are not a significant factor in the safe operation of the plant.

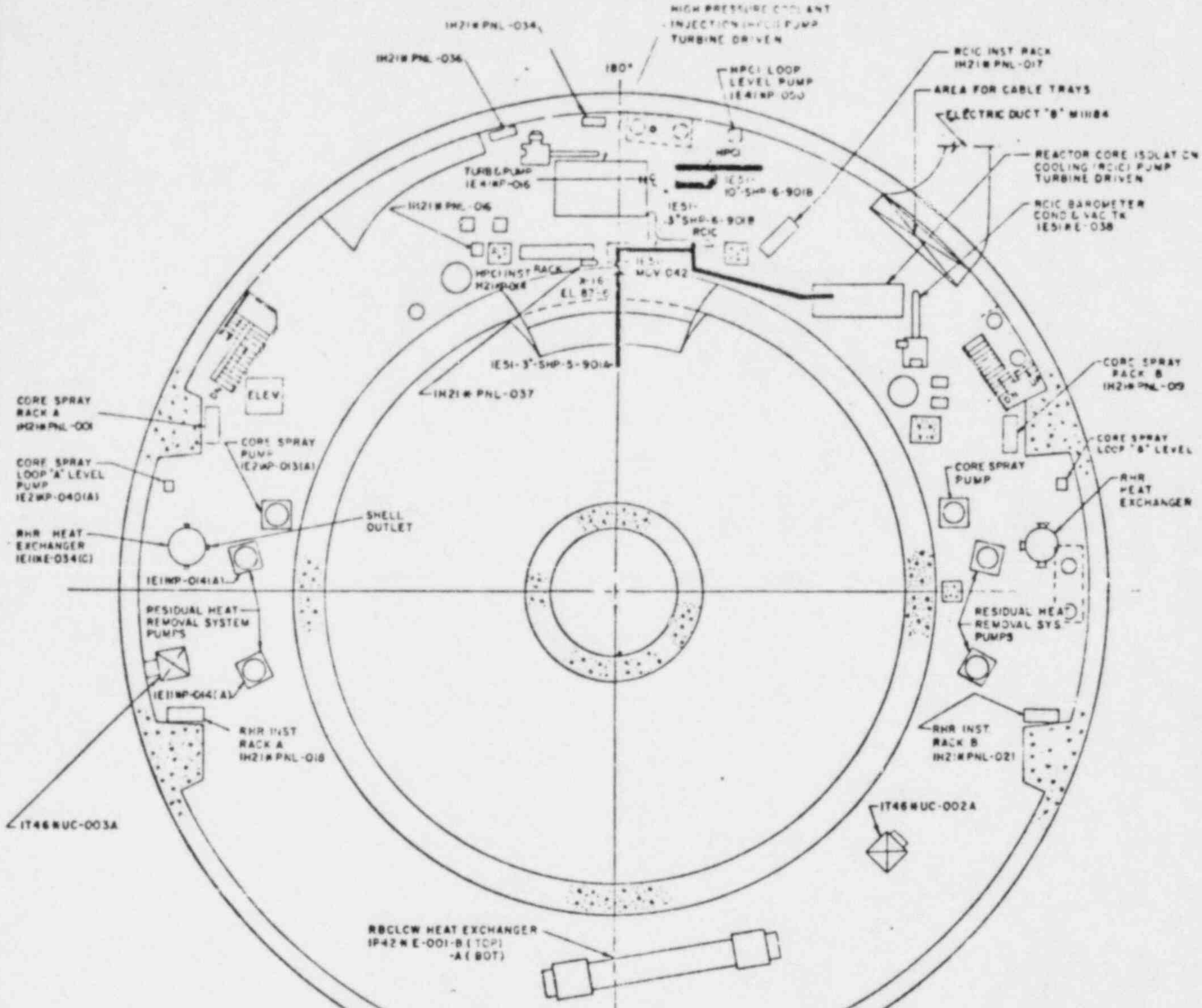
1.3 SUMMARY OF RELEVANT PLANT-SPECIFIC FEATURES OF SHOREHAM NUCLEAR POWER STATION

The Shoreham Station is a BWR-4 with a Mark II Containment. The SDV subsystem design and the secondary containment (Reactor Building) have several key plant-specific features which tend to alleviate the NRC generic concerns.

1.3.1 Secondary Containment (Reactor Building) Layout

Figures 1.1 thru 1.7 show representative plan and elevation views of the reactor building. The Hydraulic Control Units (HCUs) are on the 78'-7" elevation, while the ECCS pumps are located three floors below on the 8'-0" elevation.

On elevation 8'-0" the ECCS pumps rest on concrete pedestals and the bottom of each pump motor and turbine drive is approximately five feet above the floor. Flooding due to SDV leakage does not pose a threat to ECCS pump operability.



PLAN EL. 8'-0"

FIG. 1.1

PLAN VIEW-RELATIVE LOCATION OF HPCI AND RCIC LINES AND SAFETY RELATED EQUIPMENT

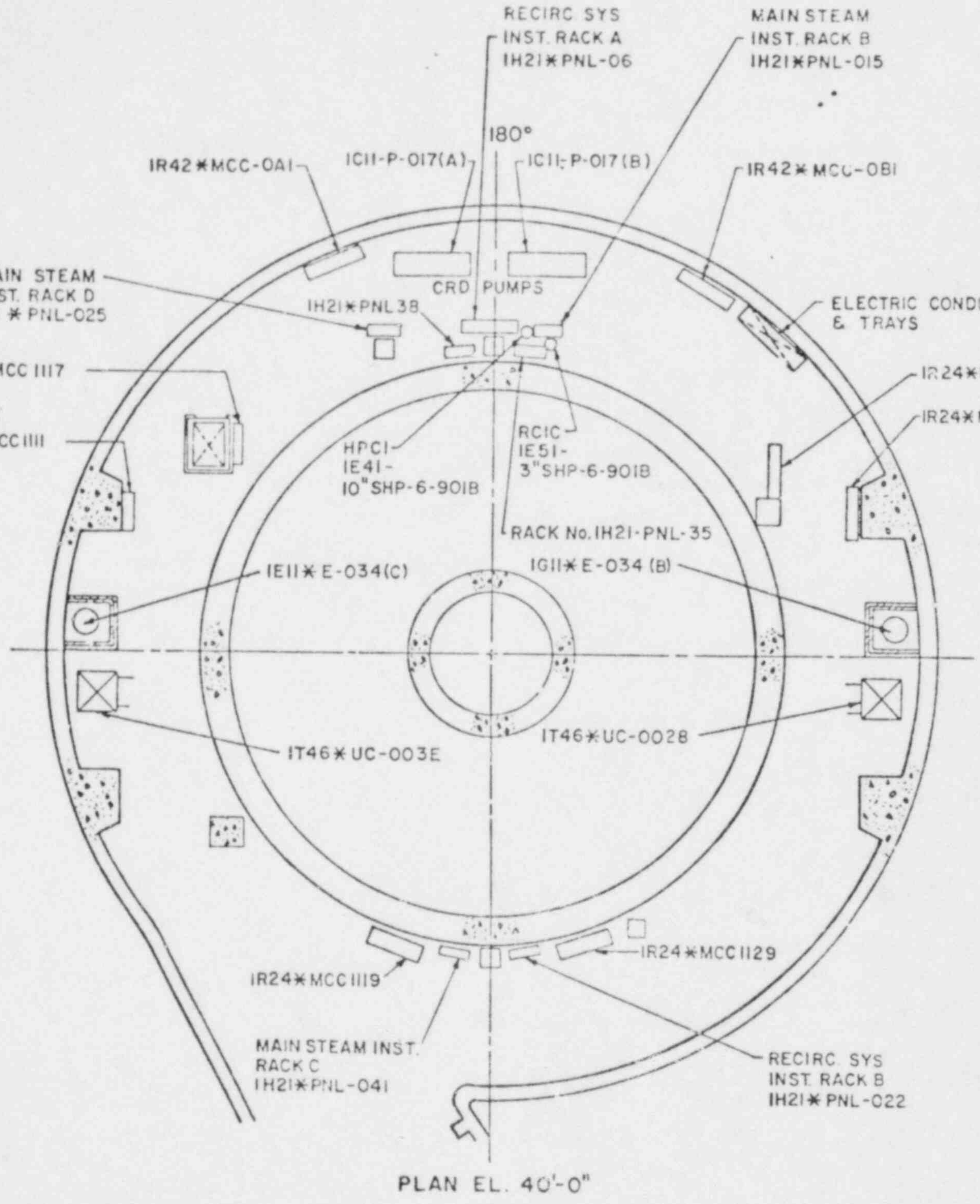
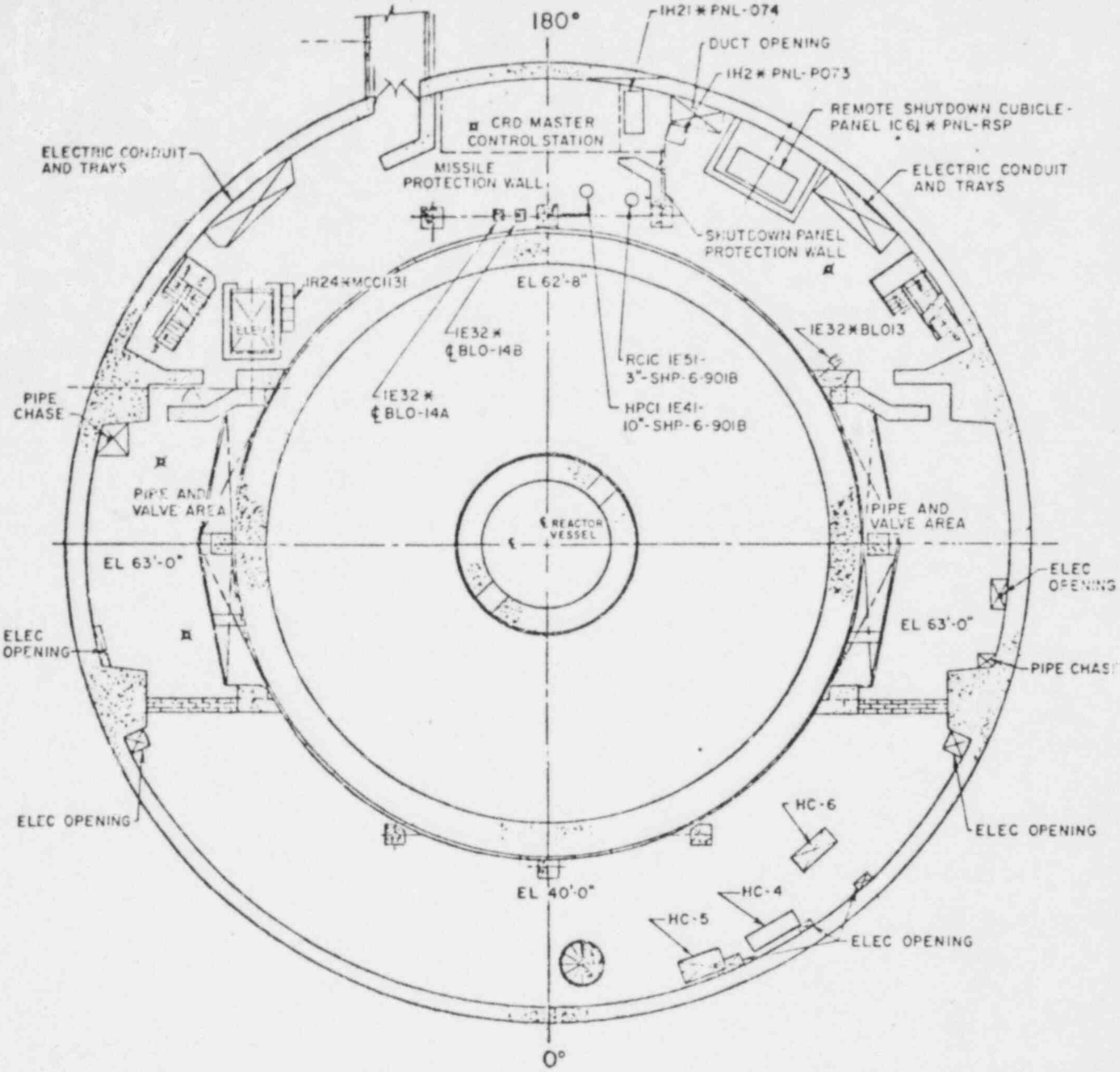
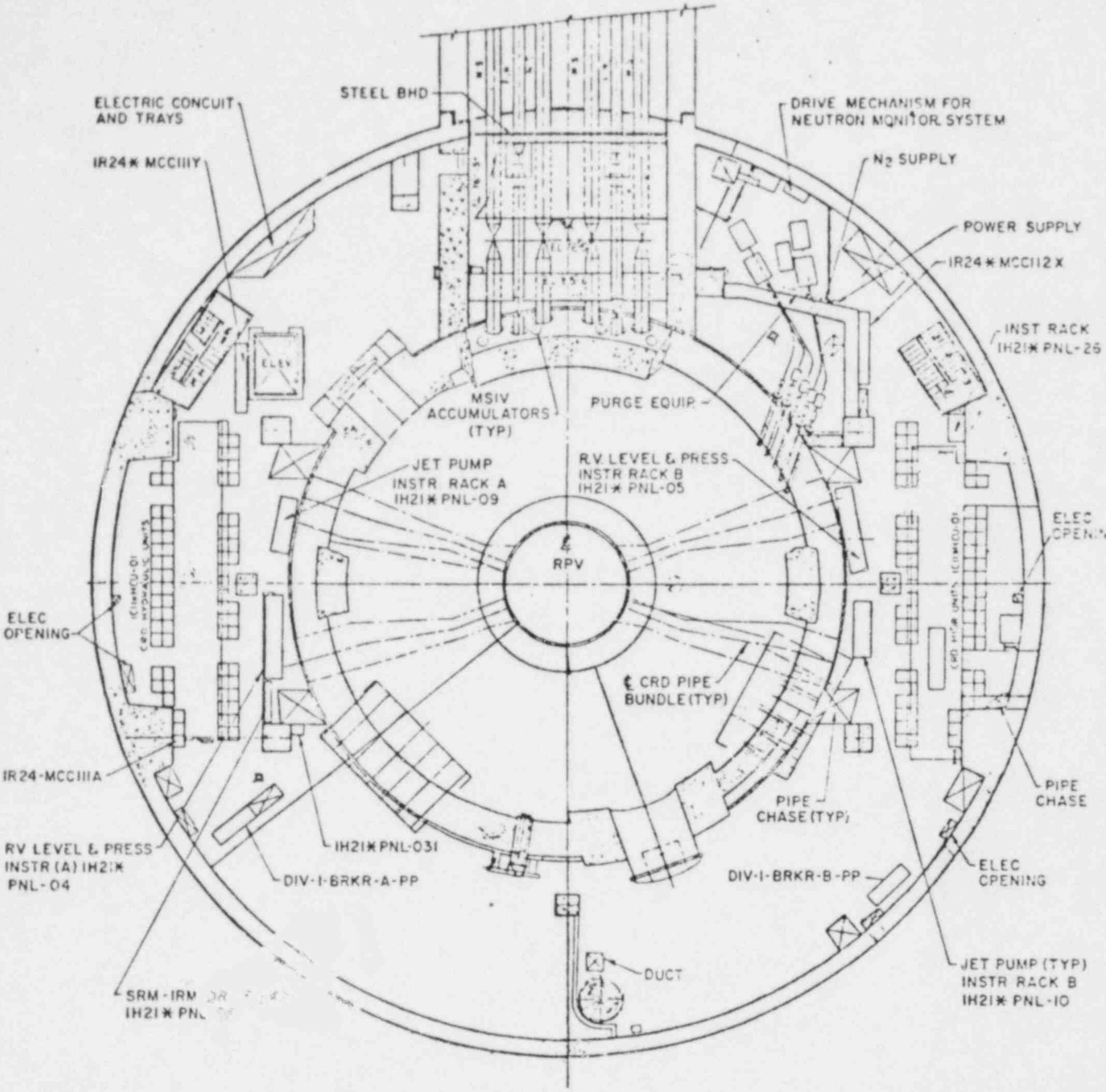


FIG. 1.2
PLAN VIEW-RELATIVE LOCATION OF
SAFETY RELATED EQUIPMENT



PLAN EL 63'-0"

FIG. 1.3
PLAN VIEW-RELATIVE LOCATION OF
SAFETY RELATED EQUIPMENT



PLAN EL 78'-7"

FIG. 1.4
PLAN VIEW - RELATIVE LOCATION OF SAFETY RELATED EQUIPMENT

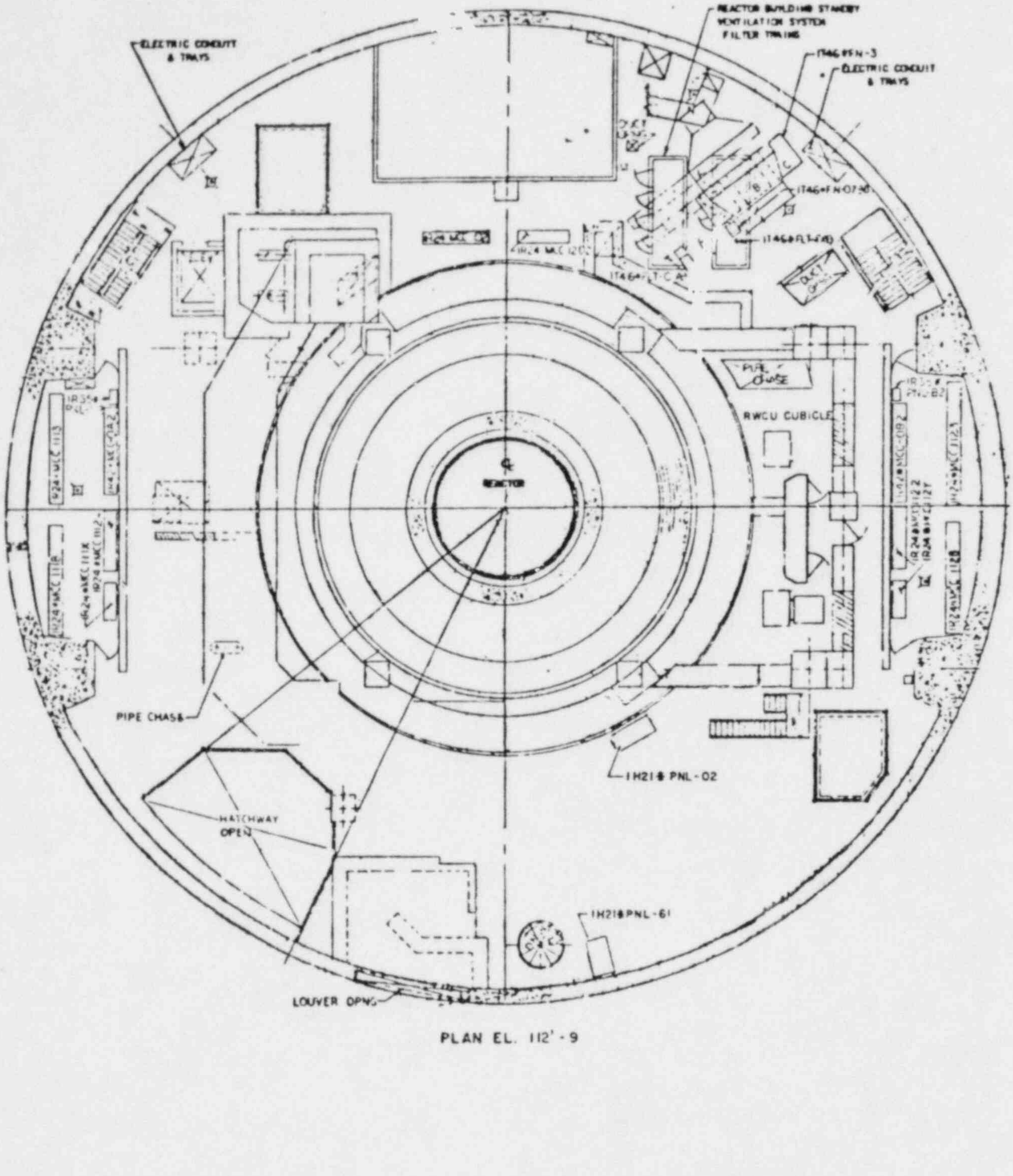
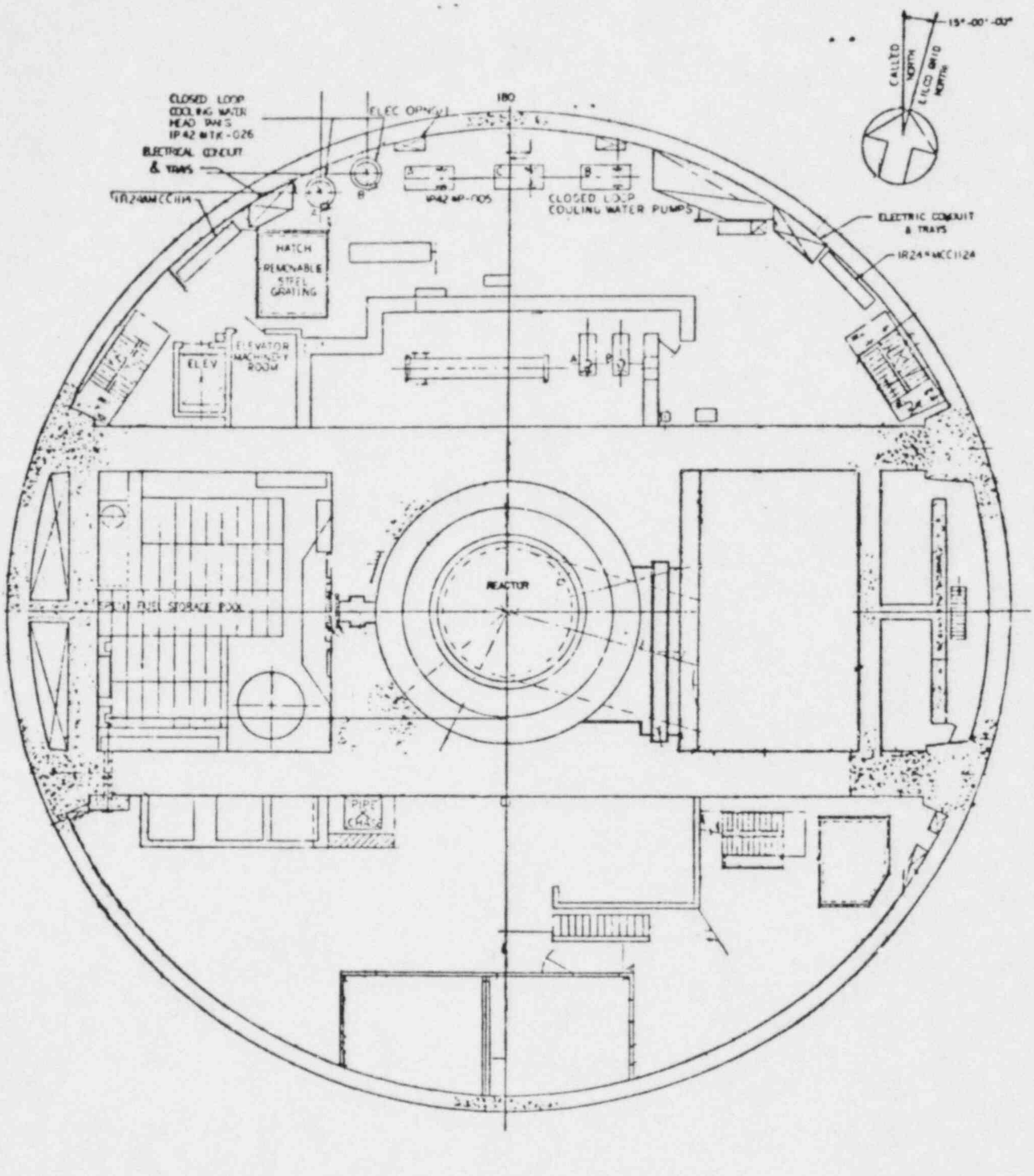
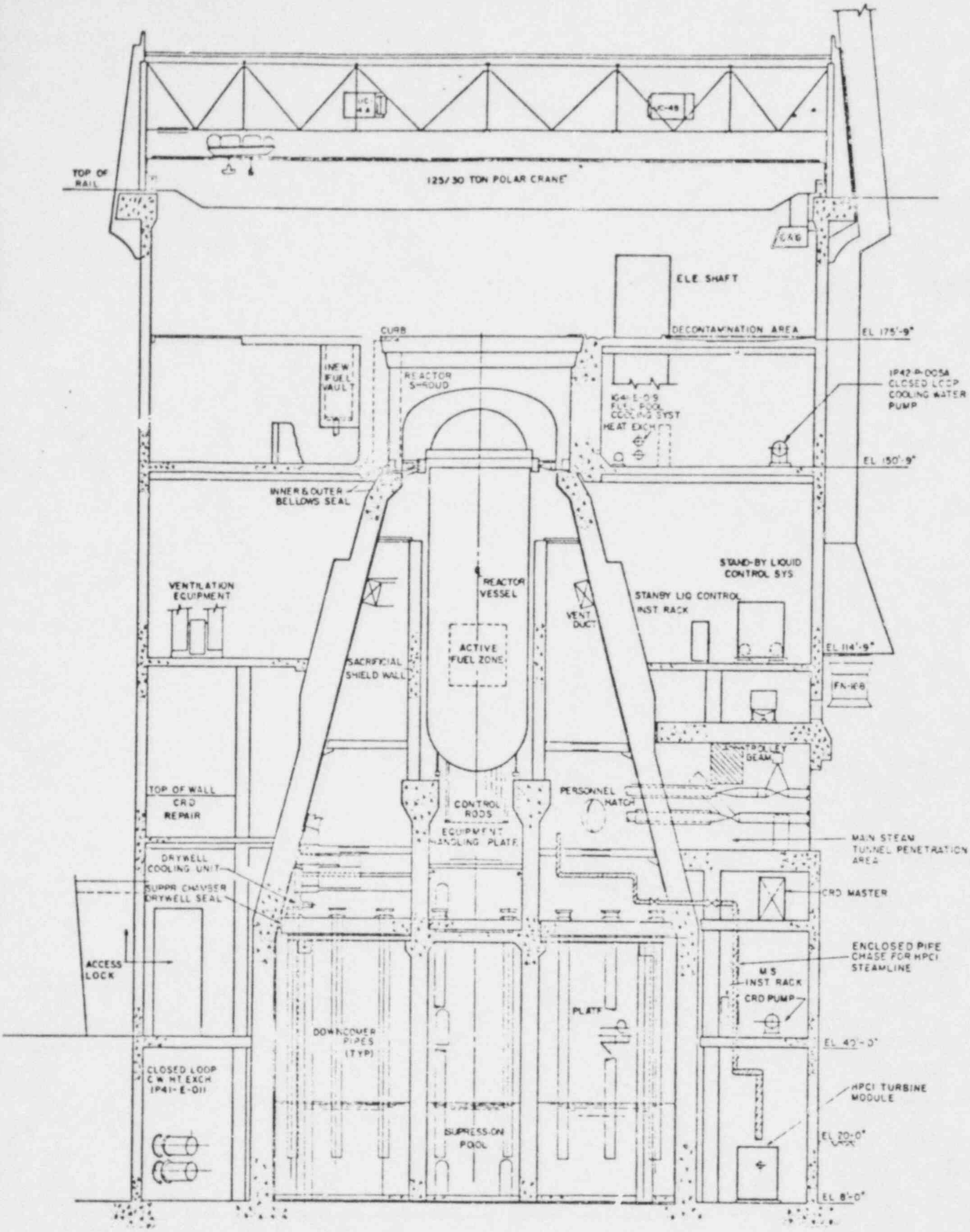


FIG. 1.5
PLAN VIEW - RELATIVE LOCATION OF
SAFETY RELATED EQUIPMENT



PLAN EL 150'-9"

FIG. 1.6
PLAIN VIEW- RELATIVE LOCATION OF SAFETY RELATED EQUIPMENT



STEAM TO HPCI TURBINE

FIG. 1.7

ELEVATION, RELATIVE LOCATION HPCI STEAM LINE TO SAFETY RELATED EQUIPMENT

1.3.2 SDV Design Features

The SNPS reactor has two banks of HCUs. Each bank has its own SDV and SDV instrument volume; these banks are separated physically on the east and west sides of the reactor. A single drain line from each instrument volume goes into a common drain line with redundant 2-inch drain valves and thence to a reactor building equipment drain sump. Separately, the vent lines from each SDV are joined at a common high point and routed via a single vent line with redundant vent valves to an atmospheric opening. Redundant drain valves and redundant vent valves have been incorporated into the SNPS design. The SDV system is designed to appropriate ASME codes and standards, has satisfied all appropriate QA procedures, and is seismically qualified.

All 3/4" piping from the HCUs to the SDV is stainless steel. In the present configuration, the 8" pipe between the HCU bays leads into a common 8" header, which in turn integrally connects to the 10" SDV instrument volume. The instrument volume empties into the 2" drain line. Throughout the remainder of this document the term "SDV system piping" or "SDV piping" will be used to describe the 3/4" piping from the HCU to the SDV header, the SDV headers and instrument volumes, and the vent and drain lines.

Each Scram Instrument Volume is monitored at three levels. At the lowest level a control room alarm is initiated, at an intermediate level the rod block is initiated to prevent rod withdrawl. Should water level reach the third and highest monitored level, diverse and redundant level instrumentation will immediately initiate a reactor scram. However, none of the Scram Instrument Volume instrumentation is necessary to mitigate the SDV pipe break event, since a scram has already occurred and other means have been identified for diagnosing the event.

1.4 PLANT-SPECIFIC CONCLUSIONS FOR SHOREHAM NUCLEAR POWER STATION

Plant-specific issues outlined in Section 1.2 have been addressed for the Shoreham Nuclear Power Station in this report. The postulated rupture of Scram Discharge Volume piping has been shown conclusively not to be a significant safety hazard

from the points of view of piping integrity, mitigation capability, or equipment qualification.

1.4.1 Fracture Mechanics Analysis of SNPS SDV System Piping

A fracture mechanics analysis of scram piping reliability was performed for SNPS in order to assess NRC issues regarding the integrity of such piping under operating reactor conditions. The fracture mechanics analysis of tolerable crack sizes was combined with estimates of the initial crack size distribution to obtain estimates of the probability of failure due to the subcritical and catastrophic growth of crack-like defects introduced during fabrication. The positive influence of the SDV system hydrostatic test was found to be large, and eliminated a number of the piping systems from consideration for failure. Fracture mechanics calculations of the remaining lines led to the following estimates (based on the most adverse weld) for the average failure rate for different size lines in the scram piping.

$<$ 2 inches	$p_f = 1 \times 10^{-7}$ per reactor year
$\geq$ 2 inches	$p_f = 9 \times 10^{-7}$ per reactor year

These values are justified to be conservative for reasons detailed in Appendix A. Comparison of the results for lines 2-inches in diameter and above with earlier estimates[1] reveal the above value to be 2-1/2 orders of magnitude lower than the earlier values. This is felt to be reasonable because the values obtained herein reflect the beneficial influence of the acceptance testing employed in these pipes and their relatively benign stress history. This is such a low probability event that it is not considered as a design basis for the plant. Nevertheless, the issues raised in NUREG-0803 have been evaluated for SNPS.

1.4.2 Piping Integrity (PI)

Appropriate quality control and quality assurance procedures both in the design and construction stages ensure that all SDV piping meets all ASME Section III Class 2 codes and conforms to the highest standards of quality.

1.4.3 Mitigation (M)

The Shoreham alarm response procedures assure early detection of any significant leak outside primary containment. If such a leak occurs and cannot be isolated, then Shoreham procedures dictate rapid depressurization of the reactor pressure vessel. Adequate core cooling capacity is available throughout any such event.

1.4.4 Equipment Qualification (EQ)

The equipment required for identifying pipe breaks outside containment,prompt reactor depressurization, short and long-term core cooling following a postulated SDV rupture with rapid depressurization, is tested in the Environmental Qualification Program to at least the Reactor Building temperatures predicted for the SDV event.

2. NUREG-0803 GUIDANCE FOR INDIVIDUAL PLANTS

2.1 INTRODUCTION

The generic response to a postulated pipe break in a BWR scram system is reported in GE report NEDO-24342. NUREG-0803 presented the NRC staff appraisal of that generic report and listed specific guidance to BWR owners for plant specific evaluations. This guidance was summarized in Table 5.2, Summary of Guidance for Individual Plants, of NUREG-0803 and is reproduced here as Table 2.1. However, the probability of SDV failure reported in the GE generic report is significantly greater than that calculated for the Shoreham Nuclear Power Station. A highly conservative fracture mechanics study shows that sufficient margin exists for a pipebreak to be a very unlikely event for the SDV piping. The scram discharge volume, the instrument volume, the vent and the drain lines have essentially zero probability of failure due to cracking. Nevertheless, each of the issues raised in NUREG-0803 is briefly discussed for SNPS in the following summary of issues. A more detailed discussion of the NRC issues is presented in Chapter. 3.

TABLE 2.1*

SUMMARY OF GUIDANCE FOR INDIVIDUAL PLANTS

Area of Concern	Guidance
PI	Periodic inservice inspection and surveillance for the SDV system
PI	Threaded joint integrity
PI	Seismic design verification
PI	HCU-SDV equipment procedures review
EQ	Environmental qualification of prompt depressurization function
PI	As-built inspection of SDV piping and supports
M	Improvement of procedures
EQ	Verification of equipment designed for water impingement
EQ	Verification of equipment qualified for wetdown by 212 ^{o}F water
EQ	Verification of feedwater and condensate system operation independent of the reactor building environment
EQ	Evaluation of availability of HPCI-LPCI turbines due to high ambient temperature trips
EQ	Verification of essential components qualified for service at 212 ^{o}F and 100% humidity
M	Limitation of coolant iodine concentration to Standard Technical Specification values

PI = Piping Integrity

M = Mitigation Capability

EQ = Equipment Qualification

* This information is reproduced from Table 5.2 of NUREG-0803

2.2 SHOREHAM NUCLEAR POWER STATION RESPONSE TO NRC GUIDANCE

2.2.1 Inservice Inspection & Surveillance of the SDV System

NRC Finding

A program of inservice inspection and surveillance for SDV piping is necessary to periodically verify its integrity.

SNPS Response

The Shoreham SDV system is comprised of ASME Section III Class 2 piping. Thus, the inservice inspection of the SDV system will be conducted in accordance with the ASME Boiler and Pressure Vessel Code Section XI, Subsection IWC. The actual ASME Section XI code used for the ISI program will be the one in effect per 10CFR50.55a twelve months prior to commercial operation. Inservice inspections will be delineated in the SNPS Inservice Inspection Program that will be submitted to the NRC for review prior to commercial operation.

2.2.2 Threaded Joints

NRC Finding

Plants with threaded joints in the SDV piping should assess their structural and leaktight integrity under conditions where severe erosion, crevice corrosion, dynamic events, and vibration can occur.

SNPS Response

There are no threaded joints in the SDV piping at Shoreham.

2.2.3 Seismic Design

NRC Finding

Each plant must verify that the SDV piping has been designed for seismic loading.

SNPS Response

All SDV system piping has had seismic loadings included in the stress analyses. The SDV system meets the requirements of ASME Code, Section III, 1971 edition and all addenda thereto, up to and including Winter 1972.

2.2.4 HCU-SDV Equipment Procedures Review

NRC Finding

The plant-specific reviews should verify that all surveillance, maintenance, inspection, or modification procedures which conceivably have the potential for defeating SDV integrity contain sufficient guidance to ensure that the loss of SDV system integrity will not occur at a time when such integrity should be available.

SNPS Response

Maintenance and mofidication work on the SDV system is controlled by LILCO Q.A. Procedures, ASME Section XI Repair/Replacement Program and Shoreham Nuclear Power Station Administrative Procedures. Work on the CRDs, HCUs, or SDV system piping would be classified as safety related or ASME Code related and would require multiple levels of review prior to commencing the work.

2.2.5 As-Built Inspection of SDV Piping and Supports

NRC Finding

The staff recommends that an as-built inspection of SDV piping and its supports be conducted at all BWRs.

SNPS Response

The as-built piping program for SNPS includes SDV piping. This program is underway and will be completed prior to fuel load (September 1982).

2.2.6 Improvement of Procedures

NRC Finding

Individual plants should implement procedures that include modifications addressing the secondary containment problems anticipated with an SDV rupture and a scram which cannot be reset.

SNPS Response

Several Shoreham Alarm Response Procedures will be modified to refer the reactor operator to the Abnormal Performance Section of the Suppression Pool Leakage Return System Operating Procedure, SP23.702.04, which describes possible reasons for the alarms anticipated following the postulated event (e.g., a leak outside primary containment) and which also contains explicit directions to the reactor operator for rapid depressurization (1) if the primary or secondary containment integrity becomes in jeopardy, (2) if the RPV level cannot be maintained, or (3) if there are excessive radiation releases to the environment.

Further, Shoreham reactor operators will receive training on how to interpret the symptoms described in this procedure to identify significant unisolable pipe breaks outside the primary containment.

2.2.7 Coolant Iodine Activity

NRC Finding

The Standard Technical Specifications (STS) for coolant activity should be implemented for all operating BWRs unless the licensee can demonstrate, based on analysis of operating history and current and projected fuel performance, that the probability of operating at coolant activity levels in excess of those allowed by STS is less than 10^{-3} per reactor year.

SNPS Response

The SNPS Technical Specification for coolant activity limits the activity to that allowed by STS.

2.2.8 Environmental Qualification for Prompt Depressurization Function

NRC Finding

The prompt depressurization function should be qualified to meet the expected SDV break environmental conditions.

SNPS Response

The prompt depressurization function is qualified to meet expected SDV break environmental conditions. The Automatic Depressurization System (ADS) consists of safety related equipment and major components are located inside primary containment. Thus, it is appropriately qualified for seismic and hydrodynamic events and major components would not be exposed to a harsh environment in the event of a SDV rupture or leak in the reactor building.

2.2.9 Equipment Qualified for Water Impingement

NRC Finding

Each licensee should verify that any emergency equipment that could be sprayed with water from dripping or splattering of overflow leakage down open stairwells is designed to operate with water impingement.

SNPS Response

In the area of the postulated pipe break on el 78'-7" there are 3 floor drains in each area which will handle part of the leakage. There is an open staircase or equipment hatchway within approximately 40' of the postulated break which water could drain down but there is no direct route for this water to impinge on the RHR or core spray pumps which are located on el 8'-0".

2.2.10 Equipment Qualified for Wetdown

NRC Finding

Licensees should verify that any equipment needed for mitigation that could be wetdown from leakage through equipment hatches is qualified for wetdown by 212 ^{o}F water.

SNPS Response

Physical location of equipment needed for mitigation is such that wetdown from leakage through equipment hatches is not considered possible.

2.2.11 Qualification of Essential Equipment

NRC Finding

Each licensee should verify that all the components of systems required for safe shutdown and long-term core cooling are qualified for service at 212 ^{o}F and 100% humidity.

SNPS Response

The equipment required to mitigate an SDV break and located within the reactor building zones as defined in the EQ program is tested in the EQ program to at least the temperatures predicted for the SDV break.

This fact has been determined by an analysis of the SNPS reactor building environment following a SDV failure.

The postulated SDV failure followed by a prompt depressurization would result in the peak zonal temperatures in six zones being above the required environmental qualification temperature previously established for the worst case pipe break outside the primary containment; however, the temperatures in the EQ testing program envelope these higher temperatures.

2.2.12 Evaluation of Availability of HPCI and RCIC Turbines

NRC Finding

Licensees should conservatively determine whether the temperature trip monitors for the HPCI and RCIC turbines would cause the turbines to trip because of temperature buildup in the areas where the sensors are located.

SNPS Response

The setpoint of the detectors is a function of those postulated breaks for which HPCI/RCIC operation is beneficial. The detectors must be set sufficiently low such that automatic isolation is effected when required. However, the setting must be above the peak calculated temperature for other high energy line breaks, such as RWCU breaks, where HPCI/RCIC could operate in the event normal feedwater were lost. However, the high pressure coolant injections systems are not required to successfully mitigate the postulated SDV event. Shoreham operational procedures will be modified such that a prompt depressurization is assured within 30 minutes after the event's initiation. If the scram prior to the SDV event causes isolations such that HPIC/RCIC are the only high pressure water sources available and the ambient temperature trips then isolate these systems prior to the prompt depressurization, Shoreham operational procedures require that an ADS be initiated immediately. In this situation the elapsed time prior to initiation of the low pressure emergency cooling systems is such that the maximum postulated leak rate will not threaten the reactor core's integrity.

2.2.13 Verification of Operability of Feedwater and Condensate System

NRC Finding

Even though the feedwater and/or condensate systems are located outside the reactor building, the licensees should verify that operation of these systems is independent of any systems or components contained in the reactor building.

SNPS Response

The feedwater and/or condensate systems are independent of any systems or components in the reactor building. The feedwater/condensate systems are not necessary to mitigate the consequences of a SDV break in our analysis, consistent with a loss of offsite power.

2.3 CONCLUSIONS

Each plant-specific requirement cited in NUREG-0803 has been incorporated or verified for the SNPS SDV system. The piping has been constructed and fabricated to ASME, Section III, Class 2. The inservice inspections will be consistent with the appropriate ASME Section XI code. The entire system is seismically supported. There are no threaded connections in the system and a final walkdown inspection will be completed by September, 1982. Further, LILCO's Q.A. and Q.C. procedures in conjunction with SNPS administrative procedures assure that the system will be operational for all necessary reactor conditions.

The SNPS operating procedures will incorporate specific directions for the reactor operator to initiate prompt depressurization when a significant unisolable leak outside primary containment is detected. The SNPS Technical Specification for coolant activity limits the coolant activity to the Standard Technical Specifications.

The equipment needed for safe-shutdown and long-term cooling is tested in the Shoreham EQ program to at least the temperatures anticipated for the SDV event reactor building environment. Water impingement and wetdown have been shown not to apply because of the SNPS equipment placement. The feedwater and condensate systems are not dependent on systems located in the reactor building.

Based upon these results, the assumptions made in NUREG-0803 for the generic integrated SDV risk assessment are fully applicable to SNPS. Therefore, the postulated SDV event does not contribute significantly to the probability of a degraded core at Shoreham.

3. SHOREHAM NUCLEAR POWER STATION EVALUATION OF SCRAM SYSTEM PIPING

3.1 INTRODUCTION

A plant specific response to NUREG-0803 requires information in three general areas - piping integrity, mitigation, and environmental qualification to support the NRC assumptions used in a generic probabilistic risk assessment which indicates that an SDV rupture following a scram which cannot be reset is not a dominant contributor to core damage or the safety of plant operation. This conclusion is further supported for SNPS by an independent fracture mechanics analysis. A detailed description of this analysis is presented in Appendix A. Specifically, it is demonstrated that the probability of an SDV failure at SNPS is sufficiently small that such a postulated event does not adversely affect the safety of the plant. The SDV hydrostatic test eliminated the possibility for failure in a number of piping systems. Fracture mechanics analysis of the remaining piping systems led to conservative estimates of the failure of the most adverse weld that are sufficiently low that failure of the BWR SDV system is not a significant contributor to either core damage or the safety of plant operation. Nevertheless, even though such a failure is extremely unlikely, the issues raised in NUREG-0803 are addressed for the SNPS in this section.

3.2 PIPING INTEGRITY

The scram discharge volume (SDV) piping system has been designed to the required ASME and ANSI applicable codes and standards. In fact, the Shoreham requirements in many cases are beyond the normal standards to reduce the system failure probability to the lowest practical value.

Shoreham piping integrity requirements are compared to industry standards in Table 3.1. Shoreham's requirements for the SDV piping system meet all of the ASME B&PV Code Section III Class 2 requirements. Details of the satisfied requirments are presented below.

TABLE 3.1

SHOREHAM SDV AND CRD PIPING SYSTEMS

COMPARISON OF SHOREHAM PIPING INTEGRITY REQUIREMENTS AND INDUSTRY STANDARDS

Piping Integrity Requirement		ASME B&PV Code Sect. III Class 2 (Requirements)	ANSI B31.1 (Requirements)	Shoreham E&PV Code Class 2 (Application)
DESIGN				
Design Specification		Yes	No	Yes
Stress Analysis		Yes	Yes	Yes
Stress Report		No	No	No
MATERIAL				
Material Specification		ASME	ASME	ASME*
Material Examination		Per Mat'l Spec	Per Mat'l Spec	Per Mat'l Spec
JOINT	Full Penetration Butt Weld	Pipe > 2"	Pipe > 3"	Pipe > 2"
	Socket Weld	Pipe ≤ 2"	Pipe ≤ 3"	Pipe ≤ 2"
	Threaded	No	Pipe ≤ 3"	No
FABRICATION AND INSTALLATION				
Process		Sect. IX plus GTAW Root	Sect. IX	Sect. IX plus GTAW Root
WELD	Weld Prep	B16.25	B16.25	B16.25
WELD EXAM	Butt Weld	RT	V	RT
	Socket Weld	PT or MT	V	PT or MT
QUALITY ASSURANCE	Qual. NDE Personal	Yes	Yes+	Yes
	Material Records	Yes	Yes+	Yes
	Fabrication Procedure	Yes	Yes+	Yes
	Fab & Exam Records	Yes	Yes+	Yes
	QA Program	Yes	Yes+	Yes

GTAW - Gas Tungsten Arc Weld
RT - Radiographic Examination
PT - Liquid Penetrant Examination
* - Material Specifications for:
Hydraulic Control Units - GE Manufacturer's Standard
MT - Magnetic Particle Examination
V - Visual Examination
\+ - Per Owner/Manufacturer Practices

3.2.1 Design

Pipe stress analysis summaries on the SDV piping system delineate the load combinations for the piping system which include seismic, hydrodynamic, thermal, and dead weight loads. The analysis of the SDV piping system has been completed to the requirements of the ASME Code, Section III, 1971 edition and all addenda thereto, up to and including Winter 1972. All the piping stresses are within the allowables and the valves have been qualified or are scheduled for qualification, as indicated below.

Seismic qualification reports on the active essential equipment in the SDV system have been reviewed by General Electric. The active essential mechanical hardware are:

1. GE MPL C11-D001, HCU, ID No. 922D249P001
2. GE MPL C11-F009, solenoid valve (instrument air)
   GE MPL C11-F182, solenoid valve (instrument air)
3. GE MPL C11-F010, 1" globe valve (vent)
   GE MPL C11-F180, 1" globe valve (vent)
4. GE MPL C11-F011, 2" globe valve (drain)
   GE MPL C11-F181, 2" globe valve (drain)

Item 1 has been qualified to the SQRT criteria (GE design Record File No. 383HA853). Items 2 - 4 are scheduled for qualification to SQRT criteria by July 1982. The SDV level switch C11N013 is qualified in General Electric file number DV159C4361.

3.2.2 Materials

Shoreham has purchased all of the SDV piping system material to the specifications in the Winter 1973 addenda of the 1971 Section III ASME Class 2 Code. Each nuclear part includes a manufacturer's data report, certificate of design specification, and certification of shop inspection. These documents are contained in the Shoreham Record Retrieval System.

3.2.3 Fabrication and Installation

All of the SDV system piping having a diameter greater than 2" was full-penetration butt-welded. This specifically includes the 8-inch scram discharge header piping and 10-inch instrument volume piping systems. These welds were performed to the requirements of ASME Section III Class 2. Construction welding inspection for these systems included radiographic examinations. Documentation is available on: (1) field and shop weld identification, (2) type of weld, (3) type of joint, (4) pipe size and schedule, (5) material specification, and (6) type of weld examination for each weld.

The SDV system piping runs having a diameter 2 inches or less were socket welded. There are no threaded connections in these systems. Each of these welds was inspected in accordance with ASME Section III Class 2 requirements using liquid penetrant (LP) or magnetic particle examination.

Since the full penetration welds were volumetrically examined and the socket welds were examined by surface methods, there is a low probability of having a fabrication defect of a size larger enough to result in a piping failure. In Appendix A, a very conservative fracture mechanics analysis of the SDV system shows that as a result of these inspections and the hydrostatic proof test the probability of a SDV piping failure due to the growth of such defects is very low.

3.2.4 Quality Assurance

Quality Assurance procedures are being used at Shoreham as illustrated in the material specification and welding certification documents described above. The Quality Assurance programs used established design and installation procedures (Ref. 2) as well as quality control inspection of installation to assure piping integrity and appropriate documentation. Nondestructive test personnel are required to be formally trained. All fabrication procedures require approval in advance.

3.2.5 Inservice Inspection & Surveillance

The Shoreham SDV system is comprised of ASME Class 2 piping. Thus, the inservice inspection of the SDV system will be conducted in accordance with the ASME Boiler and Pressure Vessel Code Section XI Subsection IWC. The actual ASME Section XI code used for the ISI program will be the one in effect per 10CFR50.55a twelve months prior to commercial operation. Inservice inspections will be delineated in the SNPS Inservice Inspection Program that will be submitted to the NRC for review prior to commercial operation.

3.2.6 As-built Inspection

An as-built inspection of the SDV and CRD piping systems and their supports will be completed at Shoreham by September 1982. As part of this program, the stress analysis will be reviewed to verify that they reflect the latest revisions of the applicable piping layout drawings (Ref. 3).

3.2.7 Maintenance and Modification of SDV Piping

Maintenance and modification work on the control rod drives, hydraulic control units, and the SDV system is controlled by LILCO QA procedures, and Shoreham Nuclear Power Station administrative procedures. Work on CRDs, HCUs, or SDVs requires multiple levels of review prior to commencing the work. Before maintenance on the CRDs, HCUs, or SDV systems can progress, review and approval of the work must be obtained from the station maintenance manager and a licensed operations supervisor to assure that technical and license committments have been met. In addition, station QA personnel review the maintenance work request for conformance to technical and quality requirements and identification of inspection hold points. Work performed under the ASME Section XI Repair/Replacement Program is also reviewed by the Authorized Nuclear Inspector. In addition to the maintenance requirements, proposed modification work must also be reviewed and approved by the station Review of Operations Committee and then forwarded to the Plant Manager for approval.

3.3 MITIGATION CAPABILITY

3.3.1 Introduction

As discussed in Section 3.2 and Appendix A a loss of integrity of the BWR Scram System Piping is extremely unlikely. Nevertheless, in consideration of a SDV failure or some other postulated pipe failure outside the primary containment at the SNPS, Long Island Lighting Company (LILCO) will modify several procedures to assure a timely and successful mitigation of the event. This section considers the operator response to a normal scram and to a postulated scram which cannot be reset followed by an SDV failure. The following topics are considered:

1. Operator training;
2. Alarms associated with the SDV event;
3. SNPS operator procedures for the activated alarms;
4. The impact of loss-of-offsite power on the operator's response; and
5. Manual isolation of the failure.

This information demonstrates that with the available leak-detection systems and the SNPS emergency procedures the reactor operators are able to determine if a significant radioactive water/steam leak exists outside primary containment such that a rapid depressurization - cooldown at a rate greater than 100 ^{o}F per hour - is necessary.

Next, this section considers the specifications for primary reactor coolant activity and the associated doses for a postulated SDV event. This is followed by an analysis of the systems available to maintain reactor pressure vessel (RPV) inventory in the event of a postulated SDV failure.

3.3.2 Operator Response to a Reactor Scram

To identify operator actions in the unlikely event of a SDV failure first consider the normal progression of events after a reactor scram. After either a manual or an automatic scram the reactor operator implements SNPS emergency procedure SP Number 29.010.01, Emergency Shutdown. Step 4.4 of SP Number 29.010.01 instructs

the operator to reset the scram. The operator can confirm that the reset is successful by monitoring the indicator lights on the full core display. The blue lights on this panel indicate that scram valves are open. Resetting the scram closes the valves and extinguishes the blue panel lights. A successful scram reset immediately terminates an SDV event. The time required to reset a scram depends on the problem which initiated the scram, but for most cases the reset would be attempted in about ten minutes or less.

3.3.3 Operator Training

All LILCO operators receive periodic simulator and classroom training. This training is designed to teach them appropriate responses for successful mitigation of any event using the Shoreham alarm response procedures, the Shoreham emergency procedures and the BWR Owner's Group Symptomatic Emergency Procedures. Thus, while LILCO does not have a specific procedure for an SDV event, the operators are trained on how to use the procedures described later in this chapter so that in the unlikely occurrance of a pipe break outside containment they would achieve a timely and successful mitigation of such an event.

3.3.4 Control Room Alarms Associated with a SDV Event

As pointed out in NEDO-24342 and NUREG-0803, there are several abnormal event signals which occur as a result of a SDV break. These include:

1. Reactor Building Area Radiation Monitor Alarm
2. Reactor Building Floor Drain Sump Level Alarm
3. CRD High Temperature Alarm
4. CRD Drift Alarm
5. Reactor Building Ventiation High Radiation Alarm
6. Reactor Building Ventilation Isolation Alarm
7. Reactor Building Differential Pressure Alarm
8. Personnel Observation of Leakage

Any one of these signals by itself would provide insufficient information on SDV status, but the combination of signals for such an event quickly identifies a significant leak outside of primary containment.

3.3.5 Alarms with Procedures that Include a Rapid Depressurization

The SNPS Reactor Building Area Radiation Monitor Alarm, the Reactor Building Floor Drain Sump Alarm, and the Reactor Building Differential Pressure Alarm response procedures will be modified to include steps which reference a procedure that describes when a rapid depressurization is warranted, i.e. the Abnormal Performance Section of the Suppression Pool Leakage Return System Operating Procedure, SP.23.702.04.

3.3.6 Operator Response to a SDV Event

The reactor operators' response to a SDV event following a scram which cannot be reset can be anticipated by examining the SNPS procedures for the alarms associated with the event. (Note, some of these procedures are not yet completed. Thus, the information presented here expresses the intent of the material to be included in these procedures. All procedures will be completed prior to fuel load.) Immediately after the scram and a postulated SDV failure, the Reactor Building Area Radiation Alarm would be initiated. This alarm response procedure directs the operator to:

- determine the affected area
- evacuate the area if necessary
- direct the Health Physics Department to validate the high radiation alarm and determine the source if possible
- refer to Abnormal Performance Section of SP.23.702.04

Thus, the operator will determine the area of high radiation and direct personnel to locate the source and refer to the Abnormal Performance Section of SP.23.702.04.

The response per station procedures for the sump alarms will provide the reactor operator with information on the extent of leakage from piping or equipment, the nature of the leak (fluid), the contamination level, the approximate size of the leak, and the affect of the leak on associated systems or vital plant equipment. Since these two station procedures reference the Abnormal Performance Section of SP.23.702.04, the reactor operator receives an indication and confirmation that a rapid depressurization in excess of the 100 ^{O}F per hour cooldown limit may be in order. Further, the operator will be referred from SP.23.702.04 to the BWR Owners' Group Procedures, Level Control and Cooldown and the Watch Engineer will have classified the event for severity and possibly initiated the Shoreham Emergency Plan. The time frame for these actions is greater than five minutes but probably less than twenty minutes. In this initial period the operator will have considerable additional information available from alarms such as the Reactor Building Ventilation High Radiation Alarm, the Reactor Building Differential Pressure Alarm, which also references SP.23.702.04, the Reactor Building Vent Supply Fan Auto Trip, Reactor Building Standby Ventilation System Alarm, Reactor Building Floor Drain Sump Excessive Run Time Alarm, and Rod Drift Alarm. Thus, the operator is aware of the relative size of the break with respect to the integrity of the secondary containment. Numerous high radiation alarms would indicate the need for immediate action to minimize radiation releases in case of potential loss of secondary containment integrity. After the operators review all the available information, the viable conclusion is that there is a break in a line with high temperature radioactive water, i.e. primary system coolant water, outside the primary containment. Further, this break either has compromised or might compromise the integrity of the secondary containment which might result in excessive radioactive releases to the environment. Because it is assumed that the operator cannot successfully reset the scram, the operator has enough information in conjunction with the directions in SP.23.702.04 to initiate a rapid depressurization and effectively terminate the accident.

3.3.7 Personnel Observation of Leakage

The reactor operators and the Watch Engineer are normally in the control room which is in the Control Building. If operating personnel are directed to, or have

other reason to enter the reactor building the most probable route would be through the Turbine Building. They would probably enter the Reactor Building at the 63' level which is one floor below the HCU-SDV area. Thus, an operator entering the reactor building in response to an alarm would immediately see some water flowing down open stairways and through open floor grates as well as the steam environment. Further, it is anticipated that the acoustic noise associated with the event will be loud, and it is likely that it would be noticed prior to entry into the Reactor Building.

3.3.8 Control Rod High Temperature Alarm (Alternate Detection of Problem)

The information cited above is sufficient for the operator to determine that a rapid depressurization is necessary in this situation. However, as pointed out by GE in response to NRC questions on NEDO-24342 for the SDV event, the Control Rod High Temperature Alarm would be activated at a greater than normal rate, e.g., GE estimates that nearly all the CRD's would exceed the high-temperature setpoint instead of the 5-15% that are usually seen after a scram. When the control rod high temperature alarm sounds the operator would refer to procedure ARP 1001. This procedure directs the operator to go to the Control Rod Drive Hydraulic Temperature Recorder, which is located in the Reactor Building on the HCU-SDV level, to determine which of the CRD's initiated the alarm. Thus, even if it is postulated that either the area radiation or the floor sump alarms failed, the operator still goes into the Reactor Building and hence would detect the problem.

3.3.9 Loss-of-Offsite Power Coincident with the SDV Failure (Alternate Detection of Problem)

The two area radiation monitors in the HCU-SDV area are not connected to an emergency power source. Thus, if offsite power is lost, these detectors would not operate. However, the el 8 reactor building high water level alarms are not dependent on offsite power. The maximum delay associated with the loss-of-offsite power would be either the time required to trip the sump or floor level alarms, or the time required for the escaping fluid to reach an active radiation monitor such as the reactor building ventilation high radiation alarm. These alarms in conjunction with operator entry into the reactor building in response to the sump

or area radiation alarms or to conditions caused by the loss-of-offsite power still assures an early detection of the problem.

3.3.10 Impact of SDV Leak Rates Less than the Maximum Postulated

If the SDV leak rate is less than the maximum postulated flow, then there may be an associated delay in alarm activation and operator response. However, if the flow rate is lower, any potential problems associated with the leakage will take a longer time to develop. Hence, the possible delayed alarms and response will not have detrimental consequences.

3.3.11 Manual Isolation of the SDV Leak

After a rapid depressurization, the flow through any SDV crack would be reduced to subcooled water so that only residual steam pockets would present high temperature problems. Further, because SNPS primary coolant activity is limited to the Standard Technical Specifications, radiation levels following depressurization would not preclude personnel entry into the HCU-SDV area. At SNPS, entry to manually isolate the leak would be made in accordance with the Shoreham Emergency Plan.

3.3.12 Primary Coolant Activity

The technical specifications for SNPS reactor coolant system currently specifies that the specific activity of the primary coolant shall be limited to less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and less than or equal to 100 / $\bar{E}$ microcuries per gram. This is the Standard Technical Specifications (STS) cited in the plant guidance of NUREG-0803, Chapter 5. Thus, the SNPS radiation levels in the reactor building would permit operator access following an SDV leak provided that routine SNPS precautions for entering potentially high-radiation areas are followed. Offsite doses would remain within regulatory limits.

3.3.13 Emergency and Long-Term Cooling Capability

In NUREG-0785, the emergency and long-term cooling of the reactor core following an SDV event was questioned on the basis that the area which houses the emergency and long-term cooling pumps would be flooded and the operability of these pumps may then be questionable. In NUREG-0803 prompt depressurization followed by manual isolation of the leak is presented as a solution which mitigates general flooding. Notwithstanding these facts, the problem of general flooding has been reviewed for Shoreham. Since the SDV crack occurs on el. 78-7, water will flow via various paths to the basement, el. 8. The actual flood depth on el. 8 would be less severe than that from a moderate (or high) energy line failure as described in FSAR Appendix 3C.4 and 3C.5. The el. 8 area is capable of storing approximately 90,000 gal. of water prior to impacting any safety related equipment. The accumulated water in the first 4 hours prior to isolation of the crack is estimated to be less than 36,000 gallons. In addition, as outlined in Appendix 3C, redundant safety grade level detection equipment exists on el. 8 which alarms when the water level exceeds 1/2". Thus, the reactor operator will be alerted of the water level in the basement. If for some reason he has not initiated prompt depressurization, this alarm would lead to detection of the problem and prompt depressurization would be initiated. Thus, flooding is relegated to a negligible concern.

3.3.14 Secondary Containment Design and Emergency Cooling Pump Locations

At Shoreham, the majority of the required ECCS equipment is located on elevation 8, the lowest Reactor Building level. Individual equipment cubicles are not utilized and the equipment is located on pedestals at various floor locations. This arrangement provides a large storage volume for postulated leakage, allowing up to 90,000 gallons of water to be contained before ECCS equipment is impacted.

Three sumps are located in the area which receive flow drainage from el 8 and other Reactor Building elevations including the elevation of the SDV. The six floor drains at el 78 would be expected to carry almost all of the SDV leakage into the el 8 area in a controlled manner. The sumps would assist in removing SDV

leakage; however, if they overflow the large storage volume would preclude the need for additional operator action and would not jeopardize ECCS equipment.

3.3.15 Emergency Core Cooling Following a SDV Failure

HPCI or RCIC could adequately maintain the RPV level during an SDV event with a maximum leakage of 411 gpm until prompt depressurization occurred. After prompt depressurization CS or any one of the LPCI subsystem pumps would be capable of adequately maintaining the RPV level.

3.3.16 Alternative Systems Available for Emergency Core Cooling

If it is postulated that the normal ECCS are not operable, then several other systems could be available to maintain RPV inventory. These include the feedwater pumps, the condensate systems and the Reactor Building service water. The availability of these systems has been discussed in NEDO-24342 and accepted in NUREG-0803.

3.4 ENVIRONMENTAL QUALIFICATION

If an SDV event is postulated, the precise conditions will depend on the availability of ventilation systems, the exact time until prompt depressurization, the magnitude of the leakage flow, and the status of the CRD pumps.

The assumption is made that the operator would depressurize the reactor 30 minutes after initiation of the accident. The zone average peak temperature in the environmental zone containing the break reaches 180 ^{O}F, which is below the previously predicted peak temperature of 190 ^{O}F for that area. The analyses also shows that the peak temperatures for most areas are below those identified by the worst case pipe break outside the containment.

The postulated SDV break peak temperatures exceed the previously predicted pipe break environmental qualification temperatures in environmental zones 2, 3, 4, 7, 18, and 20. However, the equipment required to mitigate an SDV break and located within these zones is tested in the Environmental Qualification Program to at least the temperatures predicted for the SDV break.

3.5 CONCLUSIONS

The SNPS SDV system has been designed and fabricated to ASME Section III Class 2 code. The welds have been tested according to the code by radiographic and liquid penetrant techniques. The stresses on the system have been analyzed and are within the allowables. The stress analysis included seismic loadings. LILCO Q.A. procedures and Shoreham Administrative procedures assure that the SDV system integrity will not be degraded by modification, inspection or similar work while it is required for safe reactor operation. Complementing these results is a probabilistic fracture mechanics evaluation of the SDV piping which shows that the probability of a significant fracture is less than 9×10^{-7} per reactor year.

The presently installed leak detection equipment and the SNPS procedures assure an early detection of a postulated SDV rupture and a subsequent prompt depressurization.

The reactor coolant activity limit is the STS limit. Analyses in NUREG-0803 show that for the STS limit doses to personnel entering the HCU-SDV area are not prohibitive if reasonable precautions are taken. Further, any release to the environment will be less than those specified by the current regulations.

There are numerous emergency and long-term core cooling options available to the reactor operators. The design of the secondary containment and the equipment placement are such that water impingement or flooding are extremely unlikely even if depressurization should be delayed. The equipment needed for safe shutdown after the SDV event is tested for the anticipated accident environment.

Thus, the SNPS SDV piping integrity, mitigation capability, and equipment environmental qualification assures that the postulated SDV event does not affect the safety of the Shoreham Nuclear Power Station.

REFERENCES

1. GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks, NEDO-24342, Appendix B, April 1981.

2. Quality Assurance Manual, Reactor Controls Inc., Rev. 1.

3. Shoreham Nuclear Power Station Project Procedure 42, As-built Piping Review and Reconciliation.

APPENDIX A

PROBABILISTIC FRACTURE MECHANICS
EVALUATION OF SNPS SDV SYSTEM PIPING

FRACTURE MECHANICS ANALYSIS OF SDV PIPING RELIABILITY

D.O. Harris
SAI -- Palo Alto
April 23, 1982

INTRODUCTION

In order to address concerns regarding the integrity of BWR SDV system piping (1,2), and to assess the consequences of failure of such piping, it is necessary to estimate the probability of failure of various size lines in this system. This has been accomplished in a preliminary manner (3) using procedures developed for the reactor safety study (4). However, such procedures do not take into account specifics of the pipe design and operation that are known to influence the piping integrity. Such factors include operating stress levels, number of stress cycles, and frequency of inspection and proof testing. Additionally, no estimates are made of the reliability of pipes of diameter less than 2 inches. Much of the SDV piping is 3/4 inch diameter. In order to estimate the piping reliability for a variety of pipe sizes and to account for specific operating conditions of the piping system, an analysis was performed to estimate the failure probability of the SDV piping at the Shoreham Nuclear Power Station. The procedures employed and results obtained will be presented in the following sections.

Basically, the methodology assumes that piping failures occur due to the growth of crack-like 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 as-fabricated defect distribution is altered by pre-service inspection according to detection probabilities associated with the inspection procedures. The post-inspection defect distribution then serves as initial conditions for fracture mechanics calculations of crack growth that occurs as a result of service conditions. The probability of failure at a given weld location at a given time is equal to the probability of a crack larger than the critical crack size existing at that location and time.

Such procedures are generally referred to as "probabilistic fracture mechanics" and have been widely applied to nuclear reactor pressure vessels and piping. Reference 5 provides a comprehensive review of work in this area and also serves as an example of the current state-of-the-art. References 6 - 8 provide additional discussions in this area. Figure 1 schematically shows the various steps involved in the analysis.

The SDV piping under consideration is seamless, and all welds are therefore circumferential. Interior surface part-circumferential cracks, such as shown schematically in Figure 2, are therefore the crack geometry of most concern. Complex calculations of crack growth can be performed for such cracks (5,8). The calculations can be greatly simplified if it is assumed that the crack is very much longer than it is deep ($b/a \gg 1$). In such a case the crack becomes one-dimensional and the analysis is greatly simplified. Reference 6 provides an example of a 1-D approach, which will be used in the analysis of the SDV piping system.

Hydrostatic testing, such as is performed as part of the construction acceptance test, can be very beneficial in increasing piping reliability (5,9), because the fact that a weld joint survived a hydrostatic test indicates that no cracks larger than the critical size were present during this test. This allows the crack-size distribution to be truncated at the critical size corresponding to the hydrostatic conditions. This will be discussed in more detail later, and can have a significant impact on the calculated failure probabilities.

REVIEW OF PIPING INPUTS

Pipe failures (leaks or complete pipe severances) that can produce appreciable leak rates and can not be isolated by valves are of concern. Leak rates due to failure of a 3/4 inch scram discharge line between the hydraulic control unit (HCU) and reactor pressure vessel will be limited to the leak rate past the control rod seal. Such failures are therefore not of concern. However, failure of such a line between the HCU and header can result in higher leak rates that are limited only by the 3/4 inch pipe diameter. Therefore, attention will be concentrated on all lines downstream of the HCU, including lines out to the first system isolation valve. This results in

pipe lines out to the valve in the 2 inch drain line and the valve in the 1 inch header vent being considered. Table 1 summarizes the piping subsystems considered and their corresponding sizes and wall thicknesses. Also included is the piping material and the estimated number of welds in each line. These numbers were determined either directly from piping drawings that indicate weld locations, or from piping layouts with a bend (elbow) comprising two welds, a valve two welds and a "tee" three welds. This will tend to overestimate the number of welds, because some lines are bent rather than having welded fittings. This procedure is therefore somewhat conservative.

The stresses in the various piping systems are required as inputs to the fracture mechanics analysis. The SDV piping is subjected to a number of transient types. Only normal operating stresses will be considered here, and only the stresses due to pressure, (σ_p), deadweight (σ_{DW}), and restraint of thermal expansion (σ_{TE}) will be considered. Maximum stress levels or loads for each piping system were obtained from various references with results summarized in Table 2. The axial component of the stress due to internal pressure (σ_p) was calculated from the following expression

$$\sigma_p = \frac{p\ (OD)}{4h} \tag{1}$$

where h is the pipe wall thickness and p is the reactor design pressure of 1250 psi.

The deadweight (σ_{DW}) and restraint of thermal expansion stress (σ_{TE}) were evaluated from the corresponding moments by use of the following expression

$$\sigma_{(DW\ or\ TE)} = \frac{M_{(DW\ or\ TE)}\ OD/2}{I} \tag{2}$$

The following stress components are also of interest

$$\text{load controlled stress} = \sigma_{LC} = \sigma_{DW} + \sigma_p \tag{3}$$

$$\text{cyclic stress} = \Delta\sigma = \sigma_p + \sigma_{TE} \tag{4}$$

$$\text{max. proof stress} = \sigma_{prf} = 1.25\ \sigma_p + \sigma_{DW} \tag{5}$$

The 1.25 coefficient in equation 5 is due to the pressure during the construction acceptance test being 1.25 times the operating pressure (reference 2, page 3-2).

Typically, the pressure plus deadweight stress ($\sigma_p + \sigma_{DW}$) was obtained from the sources indicated in Table 2. The sum of these stresses corresponds to equation 8 of the code employed in the initial design. Since $\sigma_{DW} + \sigma_p$ is known, and σ_p can be found from equation 1, σ_{DW} can be readily obtained. Similarly, restraint of thermal expansion stress is typically given in the sources cited in Table 2. The drain line (reference 15) is an exception to this. The values of σ_{DW} and σ_{TE} given in this reference were calculated using different stress indices than employed in the other stress analyses. This was because different versions of the code were employed due to the different years in which the analyses were performed. In order to be consistent, the detailed tabulated stresses were corrected to use the same stress indices as employed in the other lines. This consisted of using a stress index, i, of 1.3 instead of 2.1 (16).

Information on the instrumentation lines was not available at the time this report was prepared. Information in reference 11 suggests that the stresses in the instrumentation lines are less than the largest components in any of the other lines. Therefore, the values of σ_{DW} and σ_{TE} for the instrumentation lines were assumed to be equal to the largest entries for corresponding stress components shown elsewhere in Table 2. The line sizes were assumed to be the same as used in reference 11. All other stress components of interest then follow. Reference 11 indicates that stresses in lines smaller than 3/4 inch nominal diameter are so small that these lines can be omitted from consideration. The number of welds in the 3/4 and larger lines is assumed to be the same as in reference 12.

The number of times during the plant lifetime that the pipes are subjected to the stresses shown in Table 2 is also required for the fracture mechanics analysis. This number of stress cycles will equal the number of times the reactor is scrammed. Reference 1 (page A-1) suggests a rate of twice per year, which would result in 80 cycles during a 40 year plant lifetime. However, this estimate may be somewhat optimistic. Reference 3 (page A-2) suggests a rate of 6.1/year or 244 per plant lifetime, and reference 17 suggests 320 per plant lifetime. Values of 200 and 320 will be considered, with the latter value serving as an upper bound estimate.

FRACTURE MECHANICS MODEL INPUTS

The inputs to a simplified one-dimensional crack fracture mechanics model will be summarized in this section. These inputs will be combined with information provided in the previous section to perform the fracture mechanics analysis presented in the next section.

Failure Criterion: A failure criterion is required in order to define the critical crack size for the various piping systems considered. The pipes are fabricated from SA106B carbon steel, except for the 304 stainless steel scram discharge lines (See Table 1). Both of these materials are tough and ductile, and will not fail in a brittle manner. Likewise they will not fail due to a tearing instability. Reference 5 provides a detailed discussion on this topic for 304 SS. However, a catastrophic failure can occur when a crack of sufficient size (or area) exists to reduce the remaining cross-sectional area of the pipe to the point when it is not sufficient to sustain the load controlled component of the applied stress. Hence, a net section stress failure criterion is applicable. This criterion was applied to 304 SS piping in reference 5 and to SA106B carbon steel reactor piping in reference 3 and can be expressed as

$$(A_p - A_{cr})\,\sigma_{flo} = A_p\,\sigma_{LC} \qquad (6)$$

σ_{flo} is the critical net section stress, which is equal to (yield strength + tensile strength)/2. A_p is the cross-sectional area of the pipe and A_{cr} is the critical crack area. A value of σ_{flo} of 45 ksi will be used for both materials (3,5).

If the crack is taken to be one-dimensional, it can be conservatively assumed to be complete circumferential with a depth, a. Equation 6 then reduces to the following expression (see reference 6).

$$a_c = h\,(1 - \sigma_{LC}/\sigma_{flo}) \qquad (7)$$

Subcritical Crack Growth Characteristics: Subcritical crack growth in reactor piping can occur due to stress corrosion cracking (SCC), fatigue crack growth or environmentally enhanced fatigue crack growth. SCC has not

been observed in carbon steel lines, but has been observed in sensitized welds of 304 stainless steel (18). However, SCC is a time dependent process, and the scram discharge line is under load only for a short period (∿200 hrs., see reference 2). Therefore, SCC is ruled out as a significant contributor to crack growth. Environmentally enhanced fatigue crack growth then remains as the dominant contributor to subcritical crack growth. The following relations provide conservative estimates for the materials under consideration in an operating reactor environment

304 SS (ref. 19) $$\frac{da}{dn} = 10^{-9}\ (\Delta K)^4 \tag{8}$$

SA106B (ref. 20) $$\frac{da}{dn} = 1.68\times10^{-9}\ (\Delta K)^{2.37} \tag{9}$$

ΔK - cyclic stress intensity factor, ksi - $in^{\frac{1}{2}}$

da/dn - crack growth rate, inches/cycle

These relations are conservative and are applicable to high mean stresses. In fact, information in reference 5 suggests that, for stainless steel, the probability that the coefficient in equation 8 exceeds the value of 10^{-9} is 3×10^{-6}. Thus, the use of equation 8 is indeed very conservative.

Stress Intensity Factors: In keeping with the conservative use of a failure criterion for complete circumferential cracks to represent the behavior of part-circumferential cracks, the stress intensity factor relation for complete circumferential cracks subjected to uniform stress will be employed here. This relation (which is applicable for pipes with ID/h = 10) is available from reference 21 and is as follows

$$\frac{K}{\sigma a^{\frac{1}{2}}} = \frac{2+C_1\alpha+C_2\alpha^2+C_3\alpha^3+C_4\alpha^4}{(1-\alpha)^{\frac{1}{2}}} = F(\alpha) \tag{10}$$

$\alpha = a/h$ $\qquad C_3 = -6.21135$

$C_1 = -1.00250$ $\qquad C_4 = 1.79864$

$C_2 = 4.79463$

Initial Crack Distribution: The initial crack distribution consists of two components: (i) the probability of a crack existing at the weld location, and (ii) the size distribution of cracks given that a crack is present. Following

the approach of reference 5, cracks will be assumed to be Poisson distributed with a crack existence frequency per unit volume of $10^{-4}/\text{in}^3$. This value is denoted as p_v^*. The probability of having a crack in a weld of volume V, which is denoted as p^*, is then given as

$$p^* = 1 - e^{-V\, p_v^*} \tag{11}$$

The volume of weld, V, is taken to include a distance h on each side of the weld, and is given by

$$V \sim \pi(\text{ID})\, h\, (2h) = 2\pi\, (\text{ID})h^2 \tag{12}$$

The size distribution of cracks, given that a crack is present, is denoted as P_{cond}. The conditional crack depth distribution will be assumed to be exponential with a parameter, λ, of 0.246 inch. This value was used in the Marshall report (7) and was employed for the marginal distribution of crack depths in reference 5. The crack size distribution must be adjusted to account for the impossibility of having a crack deeper than the wall thickness h. The following expression for the complementary cumulative conditional crack depth distribution is obtained (5)

$$P_{cond}\,(a > x) = \begin{cases} \dfrac{e^{-x/\lambda} - e^{-h/\lambda}}{1 - e^{-h/\lambda}} & 0 \geq x \geq h \\ 0 & \text{otherwise} \end{cases} \tag{13}$$

$$\lambda = 0.246 \text{ inch}$$

This is considered to be the initial as-fabricated crack depth distribution. This distribution should be conservative for the relatively thin wall pipes under consideration, because it was estimated for reactor pressure vessels — which are much thicker.

Detection Probability: The as-fabricated crack depth distribution is modified by the detection probability of the pre-service inspection employed. The detection probability will be conservatively taken to be zero. This is equivalent to not considering the pre-service examination, and simplifies the following analysis.

FRACTURE MECHANICS ANALYSIS AND RELIABILITY RESULTS

The input components necessary to perform the fracture mechanics analysis of piping reliability have now been presented. This section will present the procedures involved and the results obtained.

Effect of Construction Acceptance Test: The hydrostatic testing performed on the piping as part of the construction acceptance test can have a strong influence on the calculated reliability. This is because the fact that the piping has survived the test means that no cracks larger than the critical size corresponding to the test conditions (a_p) existed at the time of the test — otherwise the pipe would have failed during the test. Hence, the crack size distribution can be truncated at a_p, which can have a marked effect on the calculated reliability. The following modification of equation 13 is then applicable.

$$P_{cond}\,(a>x) = \begin{cases} \dfrac{e^{-x/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} & 0 \le x \le a_p \\ 0 & \text{otherwise} \end{cases} \tag{14}$$

The value of a_p for the piping systems considered is obtainable from equation 7 with σ_{LC} taken equal to σ_{prf} (eq. 5) which is provided in Table 2. Such values will be presented along with other relevant crack sizes discussed in the next section.

Subcritical Crack Growth Calculations: The size distribution of cracks remaining after the hydrostatic test, as given in equation 14, will change during operation of the plant due to the cyclic stresses imposed. Hence, the probability of pipe failure will be time-dependent and equal to the probability of a crack larger than the critical size existing at a given time. The critical crack size is obtainable from equation 7 in conjunction with information from Table 2.

An alternative viewpoint is to consider "tolerable initial crack sizes". For a given pipe weld and time, t, this is the crack size at t = 0 that would just grow to critical size in t. Denoting this as a_{tol} (t), the probability of failure within time t is then equal to the probability of having a crack larger than a_{tol} (t) at t = 0. Hence, once a_{tol} (t) is known (which is strictly a fracture mechanics calculation) the conditional cumulative failure probability

is given by

$$P_{f(cond)}(t) = P[a > a_{tol}(t)]$$

$$= \begin{cases} \dfrac{e^{-a_{tol}(t)/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} & 0 \le a_{tol}(t) \le a_p \\ 0 & \text{otherwise} \end{cases} \qquad (15)$$

Tha value of a_{tol} (t) at t = 0 is a_c (by definition). The value of a_p will be less than a_c, because the construction acceptance test stress is higher than the load controlled stress during normal operation. Therefore, the failure probability will be zero for the time (or number of cycles) it would take a crack to grow from a_p to a_c. The number of cycles during which the failure probability will be zero is of interest, because if this number is greater than the number of stress cycles a line will see, then the line can be omitted from consideration. A conservative estimate of this number of cycles, which will be denoted as n*, can be obtained by assuming the crack growth rate (da/dn) to be equal to the value calculated using ΔK corresponding to the critical crack depth (a_c). n* is given by the following expression

$$n^* = \frac{a_c - a_p}{(da/dn)\big|_{a=a_c}} = \frac{a_c - a_p}{C\ (\Delta K)^m_{a=a_c}} = \frac{a_c - a_p}{C[\Delta\sigma a_c^{\frac{1}{2}}\ \ F(a_c/h)]^m} \qquad (16)$$

In this expression, C and m are the coefficient and exponent in the fatigue crack growth relation (equation 8 or 9) and F(a/h) is defined in equation 10. Values of n* for each piping system are summarized in Table 3. Also shown are the corresponding values of a_c, a_p and various fracture mechanics parameters.

In accordance with the above discussion, any line with $n^* > 320$ can be omitted from consideration. Hence, only the drain line and the instrumentation lines need to be considered in the remainder of this analysis. Values of a_{tol} (t) for 200 and 320 cycles of stress are required for these lines in order to determine the failure probability by use of equation 15.

The value of a_{tol} (t) can be calculated on a cycle-by-cycle basis by the following procedure.

$$a_{tol}(0) = a_c$$

$$a_{tol}(1\ \text{cycle}) = a_c - \frac{da}{dn}\Big|_{a=a_c} = a_c - C[\Delta K|_{a=a_c}]^m$$

$$a_{tol}(2\ \text{cycle}) = a_{tol}(1\ \text{cycle}) - C[\Delta K|_{a_{tol}(1\ \text{cycle})}]^m$$

$$\vdots$$

$$\text{etc.} \tag{17}$$

The appropriate values of C and m follow from equations 8 and 9. K for a given σ and a is obtained by use of equation 10 with $\Delta\sigma$ obtainable from Table 2. The results of such calculations are presented in Table 4, which also includes failure probability results discussed in the next section.

Failure Probabilities: The cumulative conditional probability of failure within n_T cycles (plant lifetime) can be calculated from the a_{tol} (n_T) results in Table 4 along with a_p and equation 15. Table 4 also summarizes the results of such calculations along with other closely related information. The conditional probability of failure of a given weld can be converted to a non-conditional value by multiplying by the probability of having a crack present in the weld (p*)

$$P_f(t) = p^* P_{f(cond)}(t) \tag{18}$$

The probability of failure in the system of L welds is then obtained by conservatively assuming that all welds in the system have the same failure probabilities but are independent of one another. The following expression provides the cumulative system failure probability within time t

$$P_{f(sys)}(t) = 1 - [1-P_f(t)]^L = 1 - [1 - p^* P_{f(cond)}(t)]^L$$

$$\sim p^*L\ P_{f(cond)}(t) \tag{19}$$

The average failure rate during the period t is obtained as

$$\overline{p}_f = P_{f(sys)}(t)/t \tag{20}$$

The time t for the 200 (or 320) scrams is taken to be the estimated plant lifetime of 40 years.

Table 4 presents the average system failure rates as estimated by the above procedures. For 320 scrams, these can be expressed as a function of pipe diameter as follows

line size, in.	average system failure rate, $Yr.^{-1}$
3/4 (small instr. line)	1×10^{-7}
2 (large instr. line and drain)	9×10^{-7}

A pipe is considered to have failed when a through-wall crack exists. Hence, both leaks and double guillotine breaks are included in the above results. Estimates of leak rate probabilities can not be obtained from these results unless some assumption is made regarding the failure mode — such as the conservative assumption that all pipe failures are sudden and complete double guillotine failures. However, such an assumption may be overly conservative, especially for the larger size lines. More complex and sophisticated analyses based on the procedures presented in reference 5 could be employed to discriminate between leaks and sudden complete severances. However, such refinements are felt to not be warranted at this time.

The above results are considered to be conservative for the following reasons:

- quantities of weld conservatively estimated,
- influence of in-service inspection ignored,
- influence of in-service proof tests ignored, only the construction acceptance test was considered,
- stress intensity factors conservatively estimated assuming all cracks to be very long relative to depth,
- initial crack depth distribution for much thicker material utilized,
- upper bound estimates on fatigue crack growth characteristics employed,
- conservative estimate of flow stress used,

- all welds in a piping system assumed to have stresses equal to those for the highest stress joint, i.e., the most adverse weld.

The failure probability for the scram system piping was estimated in NEDO-24342 (Appendix B, reference 3) to be

$$\overline{P}_f\ (\geq 2") = 3.0 \times 10^{-4}/\text{Yr}.$$

There is a 2 1/2 order of magnitude difference between this value, and the estimate derived above. The results from reference 3 were obtained by use of estimates from the reactor safety study (4) which were averaged out over many piping systems and plants. Hence, they do not reflect the hydrostatic testing and inspection schedules used in the SDV piping. Additionally, the stresses summarized in Table 2 are far below code allowables, which would tend to reduce the failure probabilities relative to lines designed more closely to code limits. Another factor is the relatively small number of stress cycles imposed on the SDV piping and the small time spent under load. Therefore, the values obtained above are felt to be reasonable and representative for the relatively low stressed limited length runs of piping employed in the SDV system.

SUMMARY AND CONCLUSIONS

A fracture mechanics analysis of SDV piping reliability was performed in order to assess concerns regarding the integrity of such piping under operating reactor conditions. The fracture mechanics analysis of tolerable crack sizes was combined with estimates of the initial crack size distribution to obtain estimates of the probability of failure due to the subcritical and catastrophic growth of crack-like defects introduced during fabrication. This mode of failure is believed to be the dominant one for the pipes considered. Environmentally enhanced fatigue crack growth was considered to be the mode of subcritical crack growth. Stress corrosion cracking (SCC) was ruled out for the carbon steel lines because such a crack growth mechanism has not been observed in this material. Additionally, SCC was not considered for the 304 stainless steel lines because of the small time they spend under stress. The influence of a construction acceptance test was found to be large, and eliminated a number of

the piping systems from consideration fc- failure. Fracture mechanics calculations of the remaining lines lec to the following estimates (based on the most adverse weld) for the average failure rate for different size lines in the scram piping.

3/4 inch	$\overline{p}_f \approx 1.0 \times 10^{-7}/Yr.$
$\geq$ 2 inches	$\overline{p}_f = 9.0 \times 10^{-7}/Yr.$

These values are believed to be conservative for reasons detailed in earlier sections. Comparison of the results for lines 2 inches in diameter and above with earlier estimates reveal the above value to be 2 1/2 orders of magnitude lower than the earlier values. This is felt to be reasonable because the values obtained herein reflect the beneficial influence of the hydrostatic testing employed in these pipes and their relatively benign stress history.

REFERENCES

1. "Safety Concerns Associated With Pipe Breaks in the BWR Scram System", NUREG-0785, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1981.

2. "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping", NUREG-0803, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1981.

3. "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks", General Electric Company, Nuclear Power System Division, Report NEDO-24342, San Jose, California, April 1981.

4. Reactor Safety Study, An Assessment of the Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission, Washington, D.C., October 1975.

5. D.O. Harris, E.Y. Lim and D.D. Dedhia, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant - Vol. 5: Probabilistic Fracture Mechanics Analysis", NUREG/CR 2189, Vol. 5, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1981.

6. D.O. Harris, "The Influence of Inspection and Crack Growth Kinetics on the Integrity of Sensitized BWR Piping Welds", EPRI NP-1163, Electric Power Research Institute, Palo Alto, California, September 1979.

7. "An Assessment of the Integrity of PWR Pressure Vessels", Report by a Study Group Chaired by W. Marshall, available from H.M. Stationary Office, London, England, October 1976.

8. D.O. Harris and E.Y. Lim, "Applications of a Fracture Mechanics Model of Structural Reliability to the Effects of Seismic Events on Reactor Piping", submitted for publication in Progress in Nuclear Energy.

9. D.O. Harris, "A Means of Assessing the Effects of Periodic Proof Testing and NDE on the Reliability of Cyclically Loaded Structures", Journal of Pressure Vessel Technology, Vol. 100, No. 7, (May 1978), pp. 150 - 157.

10. C/S Pipe Certs, supplied to LILCO by Reactor Controls, Inc., San Jose, California, no date.

11. R.M. Crawford, et al., "La Salle Evaluation Report Regarding Integrity of Scram System Piping", report to Commonwealth Edison Company by Science Applications, Inc., Oak Brook, Illinois, 1982.

12. Calculations for Deadweight and Restraint of Thermal Expansion Performed by M. Cohn, Science Applications, Inc., San Jose, California, April 1982.

13. "Pipe Stress Analysis Summary", 11600.02-AX-12A-1, Stone and Webster Engineering Corporation, LILCO Shoreham Unit 1, November 1981.

14. "Pipe Stress Analysis Summary Sheet", 11600.02-AX-12N-1, Stone and Webster Engineering Corporation, LILCO Shoreham Unit 1, November 1979.

15. "Pipe Stress Analysis Summary Sheet", 11600.02-AX-12L-1, Stone and Webster Engineering Corporation, LILCO Shoreham Unit 1, January 1980.

16. Private Communication with F. Gunnison, Science Applications, Inc., Oak Brook, Illinois, April 1982.

17. Letter from R.L. Gridley, GE - San Jose to J.H. Hannon, NRC, No. MFN-126-81, dated June 30, 1981.

18. H.H. Klepfer, et al., " Investigation of Cause of Cracking in Austenitic Stainless Steel Weldments", Report NEDO-21000, General Electric Company, Nuclear Energy Division, San Jose, California, July 1975.

19. D.A. Hale, J. Yuen and T. Gerber, "Fatigue Crack Growth in Piping and RPV Steels in Simulated BWR Environment", General Electric Company, Nuclear Energy Engineering Division, Report GEAP-24098, NRC-5, San Jose, California, January 1978.

20. M.L. Vanderglass and B. Mukherjie, "Fatigue Threshold Stress Intensity and Life Estimation of ASTM-A106B Piping Steel", American Society of Mechanical Engineers Paper 79-PVP-86, New York, New York, 1979.

21. D.O. Harris and E.Y. Lim, "Stress Intensity Factors for Complete Circumferential Interior Surface Cracks in Hollow Cylinders", Fracture Mechanics: Thirteenth Conference, American Society for Testing and Materials Special Technical Publication No. 743, Philadelphia, Pennsylvania, 1981, pp. 375 - 386.

TABLE 1

SUMMARY OF PIPING SYSTEMS CONSIDERED. SIZES AND MATERIALS FROM REFERENCE 10.

Line Name	Nom Diam	Sched	OD, in	Wall Thickness, in	Matl	No. of Welds
Scram Discharge[1]	3/4	80	1.05	0.154	304SS	NR[2]
Header	8	80	8.625	0.50	SA106B	NR
Instrument Volume	10	80	10.75	0.593	SA106B	NR
Header Vent	1	160	1.315	0.250	SA106B	NR
Drain	2	160	2.375	0.343	SA106B	40[3]
Instrumentation Lines						
Large	2	160	2.375	0.343	SA106B	11[4]
Small	3/4	160	1.05	0.154	SA106B	72[4]

(1) Material and size supplied by Long Island Lighting Company.

(2) NR means not required. See discussion in fracture mechanics analysis.

(3) Conservatively estimated by doubling the number of welds estimated in portion of drain system shown in drawing 11600.02-FP-12C-5A, Stone and Webster, 11-23-79.

(4) Estimated from results included in reference 11.

TABLE 2

SUMMARY OF STRESSES IN PIPING SYSTEMS
(all stresses in ksi)

Line Name	σ_p	σ_{DW}	σ_{TE}	σ_{LC}	σ_{proof}	$\Delta\sigma$	Source
Scram Discharge	2.13	0.14	2.02	2.27	2.80	4.15	12
Header	5.39	0.91	2.65	6.30	7.65	8.04	13
Instrument Volume	5.66	0.70	1.19	6.09	7.78	6.58	13
Header Vent	1.64	2.92	4.97	4.56	4.97	6.61	14
Drain	2.16	1.35	15.00	3.51	4.05	17.16	15
<u>Instrumentation Lines</u>							
Large	2.16	2.92	15.00	5.08	5.62	17.16	See Text
Small	1.51	2.92	15.00	4.43	4.81	16.51	See Text

TABLE 3

SUMMARY OF CRITICAL CRACK SIZES AND VALUES OF n*

Line Name	a_c, in	a_p, in	$\Delta K\|_{a=a_c}$, ksi-in$^{1/2}$	$\frac{da}{dn}\|_{a=a_c}$, in/cycle	n*, cycles
Scram Discharge	0.146	0.140	10.70	1.31×10^{-5}	488
Header	0.430	0.415	24.19	3.20×10^{-6}	4690
Instrument Volume	0.513	0.490	21.90	2.52×10^{-6}	9130
Header Vent	0.225	0.222	16.20	1.24×10^{-6}	2400
Drain	0.316	0.312	54.5	2.19×10^{-5}	182
<u>Instrumentation Lines</u>					
Large	0.3043	0.3002	46.82	1.53×10^{-5}	268
Small	0.1965	0.1947	38.00	9.32×10^{-6}	193

TABLE 4

SUMMARY OF TOLERABLE INITIAL CRACK SIZES AND FAILURE PROBABILITIES

Line Name		a_{tol}, in	Weld Vol, in^3	p*	No. of Welds	$P_{f(cond)}$ (1)	System P_f (2)	$\overline{P}_f$, yr^{-1} (3)
Drain	n_T = 200	0.318	1.25	1.25×10^{-4}	40	2.96×10^{-3}	1.48×10^{-5}	3.7×10^{-7}
	n_T = 320	0.315				7.44×10^{-3}	3.72×10^{-5}	9.30×10^{-7}
Instrumentation Lines								
Large	n_T = 200	0.3013	1.25	1.25×10^{-4}	11	0	0	0
	n_T = 320	0.2996				5.76×10^{-4}	7.92×10^{-7}	1.98×10^{-8}
Small	n_T = 200	0.1947	0.183	1.83×10^{-5}	72	0	0	0
	n_T = 320	0.1936				3.42×10^{-3}	4.51×10^{-6}	1.13×10^{-7}

(1) From equation 15, for a single joint.
(2) System $P_f \sim$(no. of welds) x (p*) x ($P_{f(cond)}$).
(3) $\overline{P}_f = P_{f(system)}/t$ (equation 20).

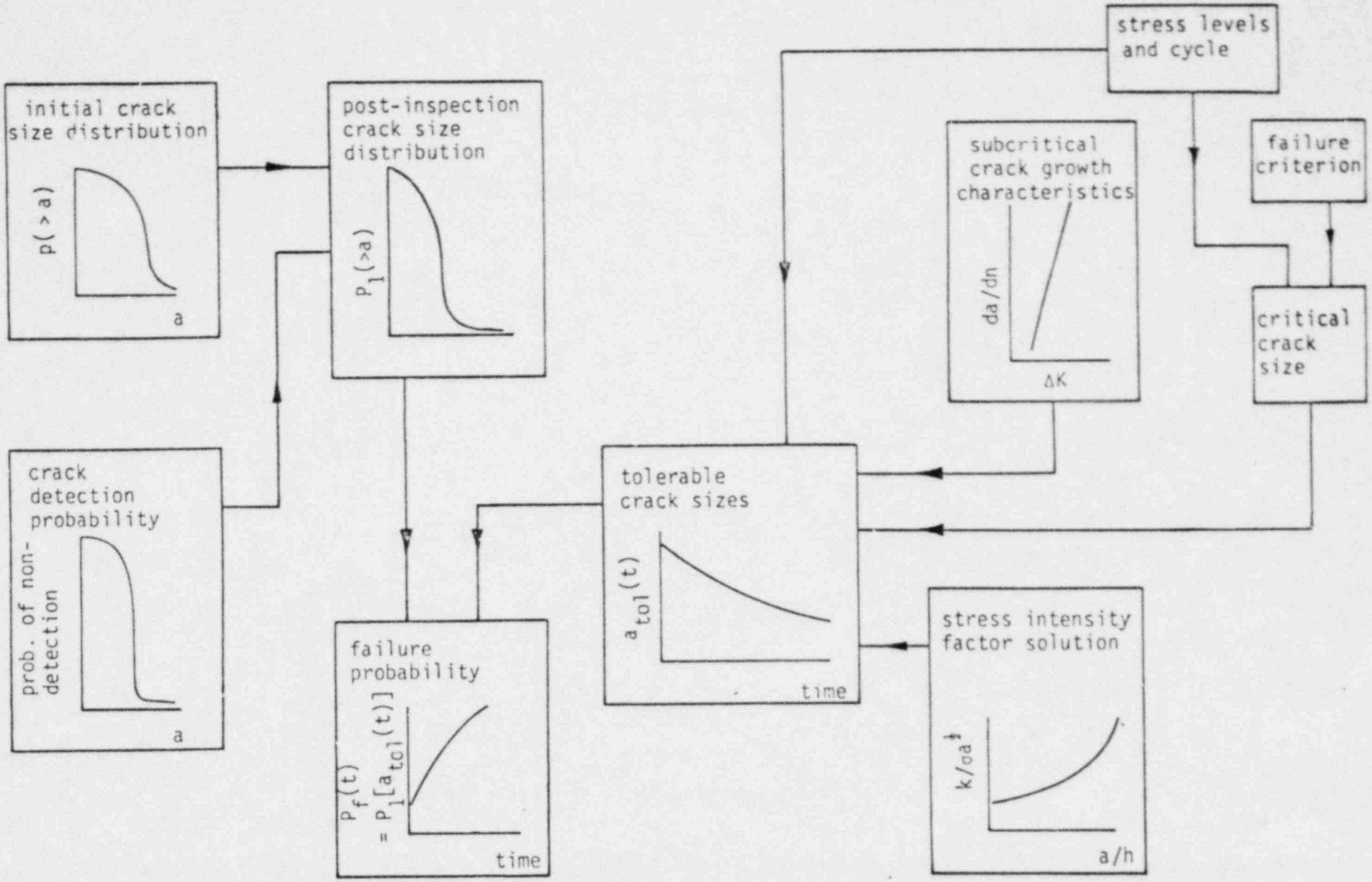


Figure 1: Schematic Representation of Various Components of Analysis of Probability of Failure of a Single Weld Joint.

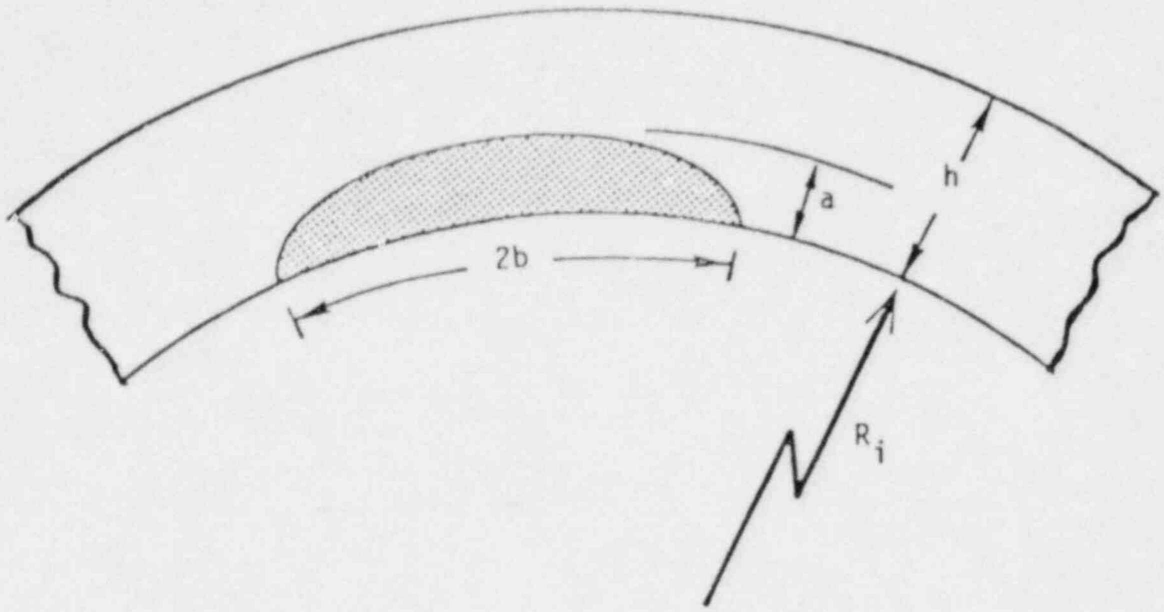


Figure 2 Geometry of Part-Circumferential Internal Surface Crack Considered in this Investigation.