REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8201040112 DOC.DATE: 81/12/29 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
50-388 Susquehanna Steam Electric Station, Unit 2, Pennsylva 05000388

AUTH.NAME AUTHOR AFFILIATION
CURTIS,N.W. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards responses to NUREG-0803, "Generic SER Re Integrity of BWR Scram Sys Piping." Complete response to QA & as-built insps will be provided prior to OL issuance.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 23
TITLE: PSAR/FSAR AMDTS and Related Correspondence

NOTES:I&E:3 copies FSAR & all amends.LPDR:2cys 05000387
I&E:3 copies FSAR & all amends.LPDR:2 cys 05000388

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
ACTION:	A/D LICENSNG		1	0	LIC BR #2 BC		1	0
	LIC BR #2 LA		1	0	PERCH,R.	01	1	1
INTERNAL:	ELD		1	0	IE	06	3	3
	IE/DEP/EPDB	35	1	1	IE/DEP/EPLB	36	3	3
	MPA		1	0	NRR/DE/CEB	11	1	1
	NRR/DE/EQB	13	3	3	NRR/DE/GB	28	2	2
	NRR/DE/HGEB	30	2	2	NRR/DE/MEB	18	1	1
	NRR/DE/MTEB	17	1	1	NRR/DE/QAB	21	1	1
	NRR/DE/SAB	24	1	1	NRR/DE/SEB	25	1	1
	NRR/DHFS/HFEB	40	1	1	NRR/DHFS/LQB	32	1	1
	NRR/DHFS/OLB	34	1	1	NRR/DHFS/PTRB	20	1	1
	NRR/DSI/AEB	26	1	1	NRR/DSI/ASB	27	1	1
	NRR/DSI/CPB	10	1	1	NRR/DSI/CSB	09	1	1
	NRR/DSI/ETSB	12	1	1	NRR/DSI/ICSB	16	1	1
	NRR/DSI/PSB	19	1	1	NRR/DSI/RAB	22	1	1
	NRR/DSI/RSB	23	1	1	NRR/DST/LGB	33	1	1
	REG FILE	04	1	1				
EXTERNAL:	ACRS	41	16	16	BNL(AMDTS ONLY)		1	1
	FEMA-REP DIV	39	1	1	LPDR	03	2	2
	NRC PDR	02	1	1	NSIC	05	1	1
	NTIS		1	1				

JAN 07 1982

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~63~~ 64 ENCL ~~58~~ 59

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101 • 215 / 770-5151

Norman W. Curtis
Vice President-Engineering & Construction-Nuclear
215 / 770-5381

December 29, 1981

RECEIVED
DEC 31 1981
US NUCLEAR REGULATORY COMMISSION
DOCUMENT MANAGEMENT BR
TIDC

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
NUREG-0803 - 120 DAY RESPONSE
SDV SYSTEM PIPING
ER 100450
PLA-987 FILE-841-2, 841-9

DOCKET NOS. 50-387
50-388

Reference: letter dated 9/14/81 from N. W. Curtis to A. Schwencer (PLA-925)

Dear Mr. Schwencer:

The attached report is PP&L's response to NUREG-0803 (Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping) for the Susquehanna Steam Electric Station (SSES). Since SSES is presently under construction, a complete response to some items listed in Table 5.1 (eg. QA and as-built inspections) cannot be provided at this time. PP&L will respond to these items before an operating license is issued.

Very truly yours,

N W Curtis

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

TEG/mjm

Attachment

B001
s
1/1

[illegible]

PENNSYLVANIA POWER & LIGHT COMPANY

RESPONSE TO

NUREG-0803

"GENERIC SAFETY EVALUATION REPORT

REGARDING INTEGRITY OF BWR SCRAM

SYSTEM PIPING"

FOR

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2

DECEMBER 29, 1981

TABLE OF CONTENTS

		PAGE
I	INTRODUCTION	2
II	BACKGROUND INFORMATION	2
III	SCRAM DISCHARGE VOLUME FAILURE SCENARIO	3
IV	RESPONSE TO GUIDANCE IN NUREG-0803 TABLE 5.1	3
V	RESPONSE TO CONCERNS NOT LISTED IN TABLE 5.1	8
VI	SUMMARY	10
VII	REFERENCES	11
VIII	FIGURES	
	Figure 1: Reactor Building Basement (El. 645'-0") . .	12
	Figure 2: Reactor Building El. 719'-0"	13
	Figure 3: Reactor Building Elevation View	14
APPENDIX A:	SDV Break Mitigation Components	A-1
APPENDIX B:	Power Supplies For Mitigation Components	B-1
APPENDIX C:	Leak Detection Devices	C-1
APPENDIX D:	SDV Break Dose Calculation Results	D-1

I. INTRODUCTION

The following report is PP&L's response to NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," for the Susquehanna Steam Electric Station (SSES). It is PP&L's intent to conform to the final approved guidance provided in Section 5, "Generic Conclusions", of the NUREG. This response will address each item listed in Table 5-1, "Summary of Guidance for Individual Plants," and provide PP&L's response on an item by item basis. Additional items which require a plant response, but are not contained in Table 5.1, will be addressed in Section IV of this report entitled, "Response to Additional Concerns".

At this time, SSES Units 1 and 2 are under construction, and therefore it will not be possible to provide a complete response to some items listed in Table 5.1 at this time. These items are mainly related to quality assurance and as-built inspections. An addendum will be provided, as necessary, to describe the follow-up work. The schedule for this addendum will be addressed in the individual section.

II. BACKGROUND INFORMATION

On April 3, 1981, the Office for Analysis and and Evaluation of Operational Data (AEOD) issued draft NUREG-0785 (Reference 1) entitled, "Safety Concerns Associated With Pipe Breaks in the BWR Scram System". This report was prepared as a result of the Browns Ferry 3 partial failure to scram of June 28, 1980. The AEOD report examined the Scram Discharge Volume (SDV) subsystem of the BWR Control Rod Drive (CRD) System with respect to specific failure conditions which were postulated to cause an eventual loss of shutdown capability. The AEOD made several recommendations for corrective action and expressed concerns regarding the integrity of the SDV piping.

After reviewing the AEOD report, the NRC Staff required all BWR licensees to perform a generic evaluation of the safety concerns expressed in the AEOD report. In response to this request, NEDO-24342, (Reference 2), "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," was submitted to the NRC on April 30, 1981. After reviewing this document, the Office of Nuclear Reactor Regulation concluded that it would be necessary to require individual plants to generically evaluate their mitigation capability for an SDV pipe break.

As a result of this review, the NRC published NUREG-0803, (Reference 3), "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping." NUREG-0803 provides the staff's guidance to individual plants for ensuring the integrity of the SDV piping system.

III. SCRAM DISCHARGE VOLUME FAILURE 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.

IV. RESPONSE TO GUIDANCE IN NUREG-0803 TABLE 5.1

PP&L's response to Table 5.1 of NUREG-0803 shall be on an item-by-item basis as follows.

1) "Provide periodic in-service inspection and surveillance for the SDV system". (Sections 2.1, 3.1.2)

   PP&L Response: The Scram Discharge Volume piping at SSES is designed to the ASME Code Section III, Class 2. Therefore, all components and their supports will be subject to the In-Service Inspection (ISI) requirements for Class 2 piping found in the ASME Section XI edition and addenda applicable to the date of issuance of Susquehanna's operating license. PP&L has revised the ISI plan for SSES to include ISI of the SDV piping, commensurate with Section XI inspection requirements for Class 2 piping. This revision will be submitted to the NRC in January, 1982.

   Also included in the ISI program will be pressure testing, which will be performed in accordance with the edition of Section XI in effect at the time Susquehanna's operating license is issued.

2) "Threaded Joint Integrity" (Section 3.1)

   PP&L Response: This item is not applicable to SSES Units 1 and 2. A review of the piping design for SSES was performed and it was confirmed that no threaded joints were used in the SDV pressure boundary.

3) "Seismic Design Verification" (Section 3.2.1.2)

PP&L Response: The SDV piping at SSES is designed as Seismic Class I piping. A design verification of the system cannot be made at this time since SSES is currently under construction. The current PP&L schedule calls for the necessary information to enable a design verification to be available by March 1982. At that time, a verification will be performed and the results will be transmitted to the NRC in an addendum to this report.

4) "HCU-SDV Equipment Procedures Review" (3.2.1.8)

PP&L Response: The primary mechanism for assuring the integrity of the SDV at SSES during operation is the administrative procedure on Equipment Status and Control. When work is to be accomplished, the maintenance organization requesting access to plant equipment must review and approve the work plan. The unit coordinator schedules the work based on requirements for current plant operating conditions and equipment status. Final approval for removal of equipment from service lies with Shift Supervision, providing assurance that essential systems, such as the SDV are not compromised. Currently the operating procedures for the CRD hydraulic system are in draft form and have not been issued. The final procedures will insure the integrity of the SDV. When these procedures are issued, a review will be performed to assure that the SDV is adequately protected.

5) "Environmental Qualification of Prompt Depressurization Function". (Section 2.3)

PP&L Response: The Safety Relief Valve (SRV) manual actuation function, which provides the prompt depressurization capability at SSES, is presently being environmentally qualified for the worst case environment which can be experienced by those components in the Containment and Reactor Buildings. Information concerning this qualification program can be found in the "Environmental Qualification Report for Class 1E Equipment," Phase II Submittal, for the Susquehanna Steam Electric Station Units 1 & 2, which was submitted to the NRC on May 1, 1981 via letter PLA-745. It should be further noted that the environmental conditions in this report meet or exceed the conditions described in NUREG-0803.

6) "As-built Inspection of SDV Piping and Supports" (Section 3.1.1)

PP&L Response: SSES Units 1 and 2 are still under ocnstruction and modifications are currently being made to the CRD system. Therefore, an as-built inspection of the SDV piping and supports cannot be made until the modifications are completed. Following

completion, an as-built inspection will be performed, the results of which will be transmitted to the NRC in an addendum to this report. The completion date for this effort is March 1982.

7) "Improvement of Procedures" (Section 4.2.3)

PP&L Response: The BWR Owner's Group will be performing a study to evaluate what modifications to the Emergency Procedure Guidelines should be made to meet the guidance of NUREG-0803. This study is currently expected to be completed near the end of the first quarter of 1982. At that time the BWR Owner's Group will determine whether to authorize specific actions to modify the Emergency Procedures Guidelines. PP&L will evaluate the specific guidelines and implement them as found to be appropriate.

In the interim period, PP&L has developed an emergency procedure for primary system leakage which may occur inside the secondary containment. This procedure (EO-00-008) adequately addresses the actions required of the operator to effectively control and/or terminate the leakage. It also describes potential symptoms, observations, automatic actions which might occur, and the actions required by the operators (including isolation and depressurization). The implications of a SDV break are also discussed in this procedure.

8) "Verification of Equipment Designed For Water Impingement". (Section 4.3.1.1)

PP&L Response: The concern expressed in NUREG-0803 on this subject involves equipment which could be sprayed with water from dripping, or splattering water overflowing the reactor building stairwells. The reactor building design at SSES consists of a Mark II type containment within a reactor building, which serves as the secondary containment. The reactor building contains each ECCS train's major components within separate water-tight compartments. Each water-tight compartment is physically remote from the stairwell, or isolated by means of water-tight, marine-type, doors. These doors are designed to withstand an average water pressure of 12 psig on one side, with a water pressure of 0 psig on the other side.

It is therefore concluded that this potential failure mode is precluded at SSES due to the structural design of SSES facilities (reference attached figures 1 & 3).

9) "Verification of Equipment Qualified for Wetdown by 212°F Water." (Section 4.3.1.1)

PP&L Response: The concern, as expressed in NUREG 0803, is that equipment hatches (to the ECCS compartments) are located on the same elevation as the SDV and HCU's. In addition, the typical reactor building sump pumps have a capacity of 50 to 100 gpm, therefore it would be expected that flooding (equipment hatch(es) open) or equipment wet-down due to drippage (equipment hatches closed) of the ECCS equipment could occur as a result of the back-up of water onto the HCU elevation (Figure 2).

The design of SSES is such that this scenario is effectively prevented. SSES is a Mark II containment design within a reactor building with water tight ECCS pump rooms (Refer to attached figures 1, 2 & 3). In addition, SSES is provided with two - 250 gpm reactor building sump pumps which can operate simultaneously and in effect will substantially reduce all aspects of flooding within the reactor building. Finally, the hatchways into the ECCS (and RCIC) rooms are located on a separate elevation from the HCU's. No significant local flooding is expected to occur on the ECCS room hatchway elevations. In addition, the Residual Heat Removal (RHR) and Core Spray (CS) room hatchways are protected by curbs and have elastomeric seals installed.

In summary, the design of SSES is such that leakage into the ECCS rooms via the hatchways is considered an insignificant problem.

10) "Verification of Feedwater and Condensate System Operation, Independent of The Reactor Building." (Section 4.3.1.3)

PP&L Response: PP&L has verified that, in the event of a break in the SDV, the operation of both the feedwater and condensate systems are independent of any systems or components contained in the reactor building. Furthermore, all of the major components of these systems are located in the turbine building and therefore will not be subjected to the harsh environment described in NUREG-0803. Additionally, the condensate pumps are electrically driven pumps and they would remain functional as long as offsite power is not lost.

11) "Evaluation of Availability of HPCI-RCIC Turbines Due to High Ambient Temperature Trips." (Section 4.3.1.3)

PP&L Response: The availability of the HPCI and RCIC systems at Susquehanna following a break or leak in the SDV piping system is a function of the increased temperatures in the respective turbine/pump rooms as a result of the postulated break. The HPCI and RCIC leak detection, area temperature instruments actuate system isolation at 167°F, and the area ventilation

differential temperature alarms at 89°F. The actuation of these high temperature instruments will result in turbine trips for the respective systems until the high temperature condition clears."

PP&L is conducting an evaluation of the expected temperature in the HPCI and RCIC rooms following the postulated break. The preliminary results of our analysis indicate that the temperature in these rooms will remain well below the trip points. Therefore, the HPCI and RCIC systems would be available for use.

12) "Verification of Essential Components Qualified for Service at 212°F and 100% Humidity (Sections 4.3.2.3 and 4.4.1).

<u>PP&L Response</u>: Appendices A, B, & C provide a list of systems and major components which may be used to detect and mitigate the effects of a break or leak in the SDV. These systems have been identified and are currently being assessed as to their environmental qualifications. There is an ongoing effort to identify any additional equipment which might be essential in mitigating the consequences of an SDV break. As this equipment is identified, it will be assessed for environmental qualification. PP&L has undertaken an effort to determine the environmental conditions (radiation, temperature, pressure, humidity) which will exist in the reactor building following an SDV break. As these conditions are finalized, it will be confirmed that the enveloping conditions of NUREG-0803 are not exceeded. These conditions will be used for the environmental qualification assessment task.

All environmental qualification efforts for those components which have not been qualified, and are essential to mitigate the leak/break, will be accomplished in accordance with PP&L's ongoing task in response to NUREG-0588 (Reference 4).

13) "Limitation of Coolant Iodine Concentration to Standard Technical Specification (STS)". (Section 4.5.1).

<u>PP&L Response</u>: Currently, the draft Technical Specifications for Susquehanna Steam Electric Station (noting that Susquehanna is in the construction phase) incorporate the STS limits for specific activities referenced in NUREG-0803. PP&L is participating in Licensing Review Group (LRG) proposals to make revisions to the STS, including a change to the reactor coolant specific activity specification. The LRG change would replace numerical activity with a requirement to maintain the offsite accident dose calculated using actual coolant activity below a specified dose level.

PP&L has completed an analysis of the offsite accident doses that would result as a consequence of a pipe break in the SDV. The summary of this analysis is presented in Appendix D.

Case A (Appendix D) represents the doses which are calculated for both in-plant and offsite cases where a constant leakage of 555 gpm is assumed for four hours. For case A a realistic assumption is made for the initial reactor coolant isotopic inventories (NUREG-0016, Reference 5).

Case B (Appendix D) represents the in-plant and offsite doses which are calculated as a result of varying the leakage rate from a maximum of 555 gpm at the accident initiation, through a gradual reduction to 73 gpm from a time of 15 minutes to 30 minutes, through a further gradual reduction to 43 gpm from a time of 30 minutes to 60 minutes. This case represents the rapid depressurization case, with depressurization initiation occurring at 15 minutes. Conservative assumptions include initial Iodine-131 Dose Equivalent concentrations of 0.2 μCi/gm at scram initiation with a full spiking factor of 500 taken for all coolant leaked, regardless of when the actual depressurization of the reactor system occurred.

In both cases pressurization of the reactor building occurs and the reactor building blow-out panels release in less than ten minutes. For this reason no credit was taken for filtration of reactor building releases in spite of the fact that the 10,000 CFM Standby Gas Treatment System (SGTS) would continue to operate and filter a portion of the release.

This completes PP&L's response to the guidance for individual plants listed in Table 5.1 of NUREG-0803.

V. Response to Concerns Not Listed in Table 5.1

1) In Section 4.3.1.2 of NUREG-0803, the statement is made that at some plants vital power is not supplied to the CRD pumps. At SSES, both CRD pumps are fed by vital 4.16 kV busses. These busses are normally fed through the ESS transformers from the startup busses. During Loss of Offsite Power (LOOP), the 4.16 kV busses would be powered by diesel-generators. The CRD pumps are classified as Non-Class 1E loads, and are automatically dumped from the 4.16 kV busses during a LOOP event. They may then be manually reloaded on the 4.16 kV busses. In addition to being supplied from high reliability power sources, the discharge of the Unit 1 and 2 CRD pumps is cross-tied such that, if required, Unit 1 CRD pumps may be used to supply hydraulic scram and cooling water to Unit 2, and vice-versa. This design results in a system with a high probability of being able to

[illegible]

[illegible]

supply cooling water to the CRD's. As pointed out in NUREG-0803, Section 4.3.1.2, the presence of cool CRD water mixing with the water discharged through the CRD seals will greatly reduce the temperature of the water ultimately discharged from the SDV. The overall effect of this design is to reduce the steam flashing at the SDV break, reduce the offsite dose consequences, and to reduce the reactor building temperature and pressure time profiles.

2) SDV Leak Detection

In NUREG-0803, Section 2.1.1, the licensees and applicants should propose a leak detection method that can provide a prompt and unambiguous notification of a leak or break in the SDV.

The instrumentation 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 RlT-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.

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.

Rod Position Indication: A large pipe break in the SDV piping system will produce 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.

Additional Leak Detection Instrumentation: The additional leak detection instruments which would be available for use are the reactor building sump level alarm, reactor building exhaust vent high radiation alarms, reactor building differential pressure indicator, ECCS and RCIC pump room level alarms, and loss of reactor building ventilation alarms.

VI) SUMMARY

It is PP&L's intent to respond to NUREG-0803 by complying with the guidance given in table 5.1 of the NUREG. PP&L believes that there are several aspects of the response which should be highlighted, since they demonstrate PP&L's effort to reduce both the possibility and severity of a SDV break or leak. PP&L has developed an operating procedure which specifically includes a break in the SDV and provides for depressurization and visual observation. In the area of leak detection PP&L believes that it is possible to provide the operator with prompt detection of a break through the CRD area radiation monitors, the CRD temperature monitors, and the control rod position indication system. The design of the reactor building at SSES also provides additional protection to the ECCS equipment through the water tight compartments and the location of the equipment hatches below the HCU/SDV floor level. Furthermore, the location of the major components of the condensate, feedwater and the CRD pumps outside of the reactor building provides additional protected mitigation capability.

Since SSES is under construction, the following activities will be addressed in the future: seismic design verification of the SDV by walkdown; on-going environmental qualification of mitigation/detection components; as-built inspection of the SDV. PP&L will evaluate the results of the BWR Owner's Group evaluation of what modifications to the Emergency Procedure Guidelines should be made to meet the guidance of NUREG-0803. As these items are completed, PP&L will inform the NRC and issue them as addendums to this report.

VII) REFERENCES

1. U.S. Nuclear Regulatory Commission, "Safety Concerns Associated with Pipe Breaks in the BWR Scram Systems", USNRC Draft Report NUREG-0785, April 1981.

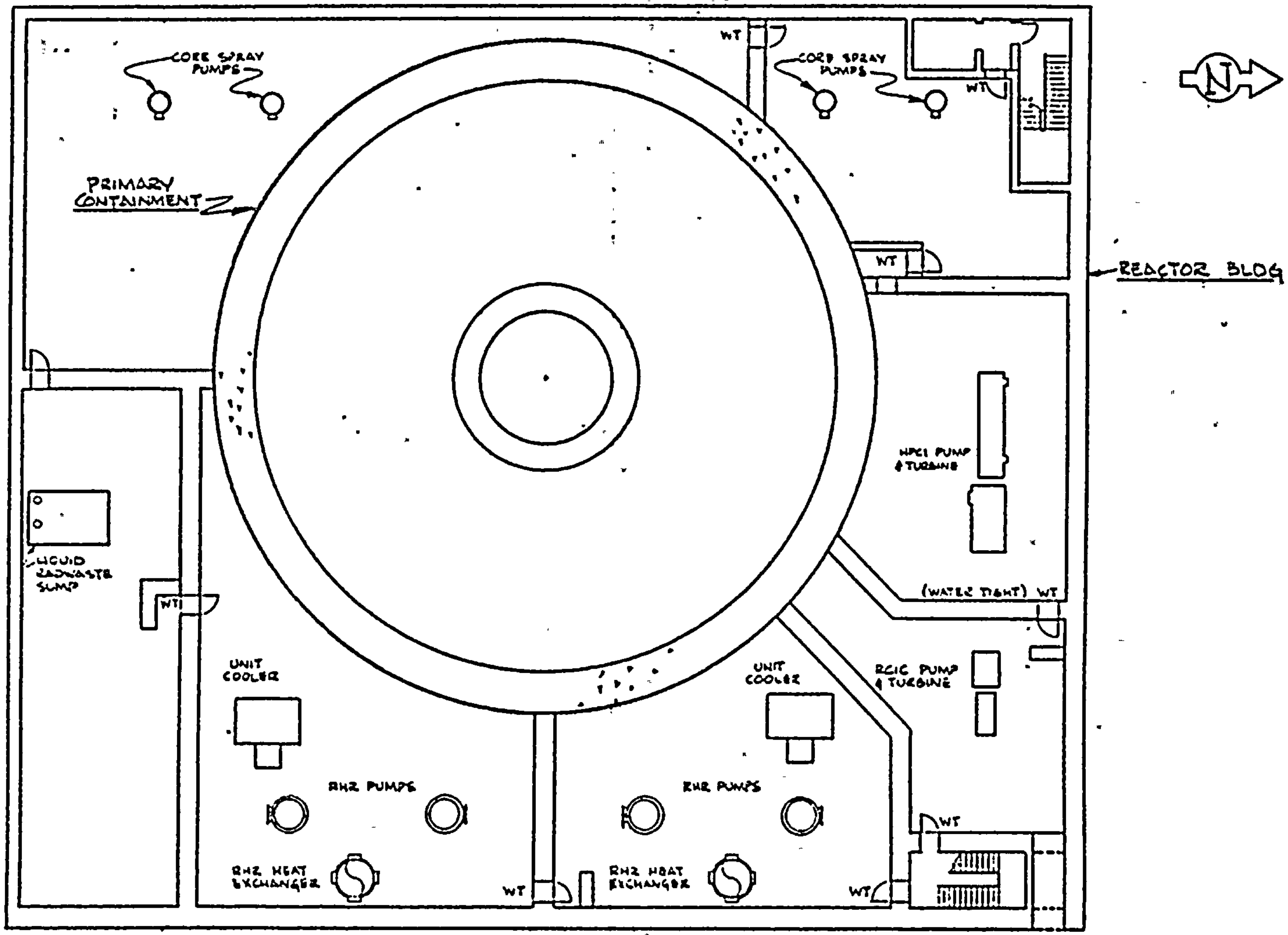
2. General Electric Company, "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks", NEDO-24342, April 1981.

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

4. U.S. Nuclear Regulatory Commission, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", NUREG-0588, Rev. 1, July 1981.

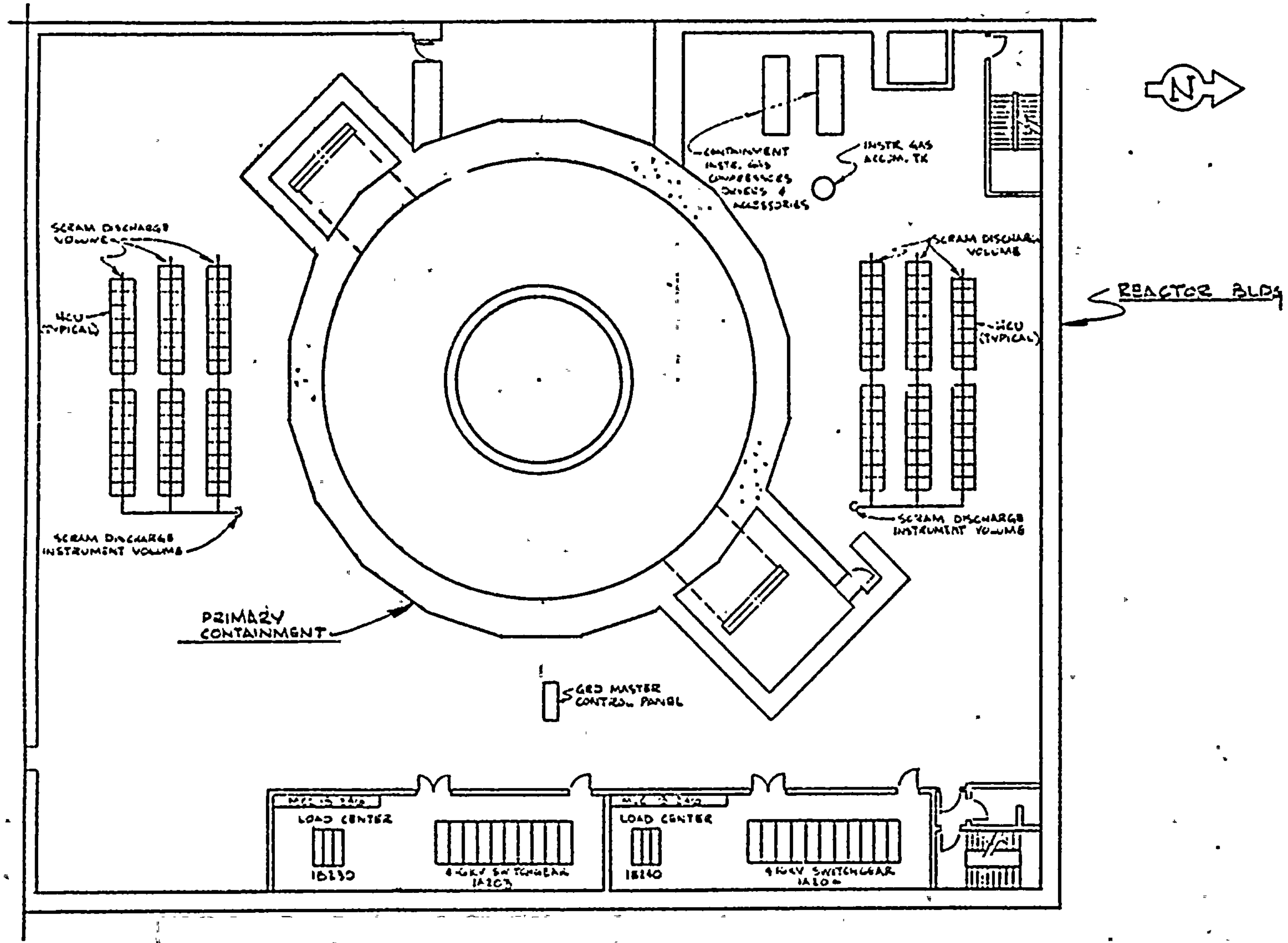
5. U.S. Nuclear Regulatory Commission, "Calculation of Release of Radiation Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)", NUREG-0016, Rev. 1, January 1979.

(MG/34-A)



BASEMENT PLAN - EL. 645'-0"

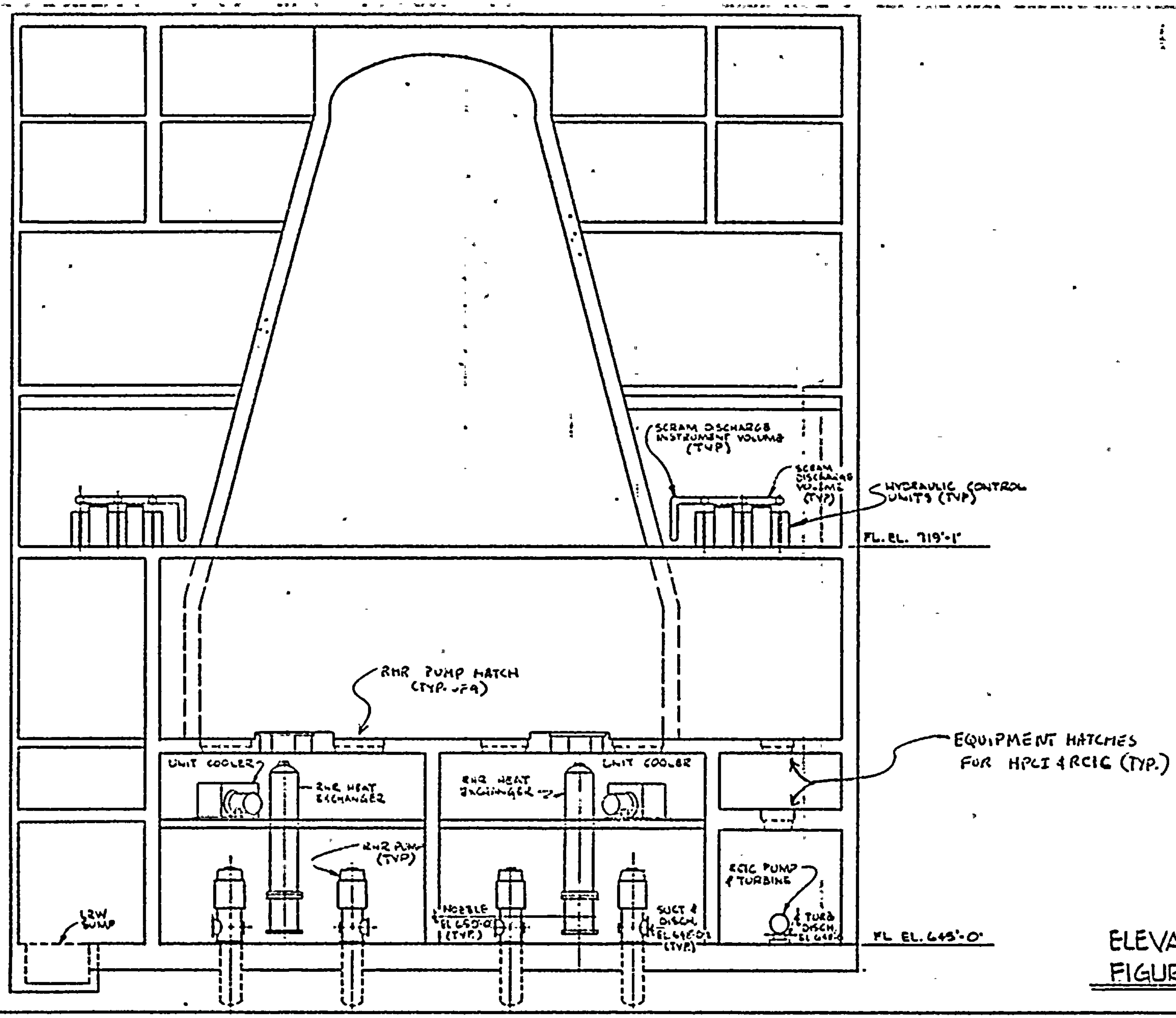
FIGURE 1



FLOOR PLAN - EL. 719'-1"

FIGURE 2

[illegible]



ELEVATION
FIGURE 3

APPENDIX A

List of components which could be used to mitigate an SDV leak or break.

I) High Pressure Coolant Injection System (HPCI).

-HPCI Pump 1P-204
-HPCI Booster Pump 1P-209
-HPCI Vacuum Tank Condensate Pump 1P-215
-HPCI Barometric Condenser Vacuum Pump 1P-216
-HPCI Turbine 1S-211
-Valves: as required to operate the system

II) Reactor Core Isolation Cooling System (RCIC)

-RCIC Pump Turbine 1S-212
-RCIC Vacuum Tank Condensate Pump 1P-220
-RCIC Turbine Lube Oil Cooler 1E-212
-RCIC Barometric Cond. Vacuum Pump 1P-219
-RCIC Turbine Control Logic
-Valves: As required to operate the system

III) Core Spray System

-Core Spray Pumps 1P-206 A, B, C, D
-Valves: As required to operate the systems

IV) Residual Heat Removal System (RHR)

-RHR Pumps 1P-202 A, B, C, D
-RHR Heat Exchanger 1E-205 A, B
-Valves: As required to operate the system

V) Control Rod Drive System

-Reactor Protection System (Portions within Reactor Bldg.)
-Reactor Manual Control System (Portions within Reactor Bldg.)
-Valves: As required to operate the system

VI) Nuclear Boiler Vessel Instrumentation

-All instrumentation associated with the Automatic and Manual Depressurization System.

Instruments

LIS 1N024 A-D
LITS 1N026 A-D
LIS 1N031 A-D

LIS 1N042 A, B
LITS 1N037 A, B
LIS 1N025 A-D
PS 1N020 A-D
PS 1N021 A, C
PIS 1N021 B, D
PS 1N022 A-S
PS 1N023 A-D
PS 1N025 A-D
PS 1N015 A-D
PSH 1N056 A-D
PT 1N055 A, B
LT 1N027
PDT 1N032
PS 1N021 E, G

VII) Area Radiation Monitoring and Meteorological Instruments

Instruments

RE-13701
RE-13702
RE-13703
RE-13704
RE-13705
RE-13706
RE-13712
RE-13714

VIII) Emergency Service Water

-Core Spray Pump Room Unit Coolers 1V-211 A-D
-HPCI Pump Room Unit Coolers 1V-209 A, B
-RCIC Pump Room Unit Coolers 1V-208 A, B
-Emergency Switchgear Load Center Room Cooler 1V-222 A&B
-RHR Pump Room Unit Cooler 1V-210 A-D
-RHR Pump Motor Oil Cooler 1E217 A-D
-RHR Pump Seal Water Cooler 1F-218 A-D
-Valves: As required to operate the system

IX) RHR Service Water

Valves: As required to operate the system

X) Containment Instrument Gas

-Valves: As required to operate the system

APPENDIX B

List of Power Supplies for mitigation components.

I) High Pressure Coolant Injection System (HPCI)

1D624	Class 1E
1D264	Class 1E
1D274	Class 1E
1D254	Class 1E
1D2640	Class 1E
1D2640	Class 1E

II) Reactor Core Isolation Cooling (RCIC)

1D2500	Class 1E
1D614	Class 1E
1D254	Class 1E
1D624	Class 1E
1D264	Class 1E

III) Core Spray System

1A2010	Class 1E
1A2020	Class 1E
1A2030	Class 1E
1A2040	Class 1E
1B216	Class 1E
1B226	Class 1E
1B217	Class 1E
1B227	Class 1E
1B237	Class 1E
1B247	Class 1E

IV) Residual Heat Removal System (RHR)

1A2010	Class 1E
1A2020	Class 1E
1A2030	Class 1E
1A2040	Class 1E
1B216	Class 1E
1B226	Class 1E
1B237	Class 1E
1B247	Class 1E
1B246	Class 1E
1Y216	Class 1E
1Y226	Class 1E
1D274	Class 1E

1B236	Class 1E
1B219	Class 1E
1B229	Class 1E
1B217	Class 1E

V) Control Rod Drive System

1Y218	Non-Class 1E
1Y624	Non-Class 1E

VI) Nuclear Boiler Vessel Instrumentation

1Y201A	Non-Class 1E
1Y201B	Non-Class 1E
1D614	Class 1E
1D624	Class 1E
1D614 (A-S)	Class 1E

VII) (Area Radiation Monitoring and Meteorological Instruments

1LP21A-MCC1B252	Non-Class 1E
1LP22A-MCC1B253	
1LP24A-MCC1B271	

VIII) Emergency Service Water

1B216	Class 1E
1B226	Class 1E
1B237	Class 1E
1B247	Class 1E
1B227	Class 1E
1B217	Class 1E

IX) RHR Service Water

1B237	Class 1E
1B247	Class 1E
1C617 (1Y216)	Class 1E
1C618 (1Y22)	Class 1E

X) Containment Instrument Gas

1B217	Class 1E
1B236	Class 1E
1Y216	Class 1E
1D614	Class 1E
1Y236	Class 1E
1Y246	Class 1E
1Y226	Class 1E

APPENDIX C

List of Leak Detection Devices:

- Temperature element in the RPIS probes.

- Area Radiation Monitors:
  R1T - 13705
  R1T - 13606

  - Control Rod Position Indicators

(MG/34-A)

APPENDIX D

SCRAM DISCHARGE VOLUME PIPE BREAK
DOSE CONSEQUENCES

	Case A	Case B
IN PLANT		
Whole Body Dose Rate (Rem/hr)		
@ 2 Hr	1.7	0.52
@ 4 Hr	2.1	0.74
Thyroid Dose Rate (Rem/hr)		
@ 2 Hr	362	1500
@ 4 Hr	663	3000
OFFSITE		
Whole Body Dose (Rem)		
2 Hr EAB*	0.25	0.11
Total Release LPZ**	0.014	0.009
Thyroid Dose (Rem)		
2 Hr EAB	3.6	18
Total Release LPZ	0.26	2.9

* Exclusion Area Boundry
** Low Population Zone

TABLE D-1

KEY PARAMETERS FOR
SDV PIPE BREAK DOSE
ASSESSMENT

PARAMETER	CASE A	CASE B
o Radioisotopes	D.E. I-131 Kr, Xe	D.E. I-131 Kr, Xe
o Percent Coolant Flashed	45	45
o Partition Factor	0.1	0.1
o Plateout Factor	0.5	0.5
o Rx Bldg. Volume (ft^3)	1.56×10^6	1.56×10^6
o Mixing in Rx Bldg.	50%	50%
o Rx Bldg. Exhaust (CFM)	10,000	10,000
o Initial Coolant Activity I-131 Dose Equiv. (μCi/gm)	NUREG-0016 + Noble Gases	0.2 + Noble Gases
o Coolant Leak Rate (gpm)	555	0-15 min. @ 555 15-30 min. @ 314 30-60 min. @ 58 60-240 min. @ 43
o Iodine Spike	500	500
o Spike Duration (Hrs)	4	4

(MG/34-A)

TABLE D-2