UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC	)	Docket Nos. 50-440
ILLUMINATING COMPANY, et al	)	50-441
	)	
(Perry Nuclear Power Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF MONTY A. ROSS
IN SUPPORT OF NRC STAFF'S MOTION
FOR SUMMARY DISPOSITION OF ISSUE NO. 5

Monty A. Ross, being duly sworn, deposes and says as follows:

1. I, Monty A. Ross, am a manager in the Plant System Design organization of the General Electric Company. My business address is 175 Curtner Avenue, San Jose, California 95125. A summary of my professional qualifications and experience is attached hereto as Exhibit "A". I have personal knowledge of the matters set forth herein and believe them to be true and correct.

2. I have reviewed the NRC Staff's Motion for Summary Disposition of Issue No. 5, dated November 9, 1982, and supporting documents, including the Affidavit of Nicholas E. Fioravante in Support of Summary Disposition of Issue No. 5. I agree with the statements contained therein and give this affidavit in support of the Staff's motion.

3. Issue #5 references the NRC draft report NUREG-0785 and contends that the, "applicant has not demonstrated the safety of its reactor from an unrecoverable loss of coolant accident (LOCA) which could occur from a pipe break in the scram discharge volume". The referenced draft report is entitled, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System" and describes a postulated sequence of events in which a break occurs in the scram discharge volume (SDV) piping of a General Electric Boiling Water Reactor. The report assumes that the break cannot be isolated, and that the coolant leaving the reactor vessel through the break can flood or otherwise disable all emergency core cooling system (ECCS) pumps. The presumed loss of make-up water to the reactor vessel is further assumed to result in the fuel no longer being covered and cooled by water.

The Perry Nuclear Power Plant (PNPP) is a General Electric BWR/6 plant with a Mark III type containment. The SDV piping is a BWR/6 Mark III plant is located entirely inside the primary containment and directly above the suppression pool. The normal coolant makeup system and the ECCS are located outside the primary containment in the Auxiliary Building, and would not be affected by a postulated SDV rupture. In addition, since the SDV piping is located directly above the suppression pool, water discharged from a postulated SDV rupture would fall through the open steel grating of the floor below the SDV and into the suppression pool. Since the suppression pool is a primary source of water for the ECCS, the water discharged from a postulated SDV rupture becomes available for delivery back to the reactor vessel so that a closed flow loop is maintained which assures long-term core cooling capability.

Therefore, a postulated SDV rupture at the Perry Nuclear Power Plant does not present the threat to the long term core cooling capability postulated in NUREG-0785. Furthermore, the NRC staff concluded in their generic safety evaluation report on this subject of BWR scram system piping integrity, NUREG-0803, that a postulated SDV rupture for a BWR/6 Mark III containment design presents no threat to the long-term cooling capability provided by the ECCS.

The PNPP scram discharge system, which is depicted in Figure 1, consists of 177 Control Rod Drives (CRD) and the 177 associated CRD withdraw lines, the scram discharge volume and the SDV vent and drain valves.

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

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

When the scram signal is "reset" by the operator, the scram discharge valves close and the scram discharge volume vent and drain valves open, allowing the scram discharge volume to empty and return to atmospheric pressure for normal operation.

The entire scram discharge system must function reliably at full reactor pressure for short time periods following each scram. Therefore, the entire scram discharge system is designed to quality standards which reflect the importance of its occasional connection with the reactor coolant pressure boundary. All piping in the scram discharge system at Perry Nuclear Power Plant, is designed in accordance with the requirements in Section III of the "Boiler and Pressure Vessel Code" of the American Society of Mechanical Engineers (ASME Code) for Class 2 components. The scram discharge system piping is evaluated as Seismic Category 1 piping. The result of all these requirements is a high integrity system of a quality fully in

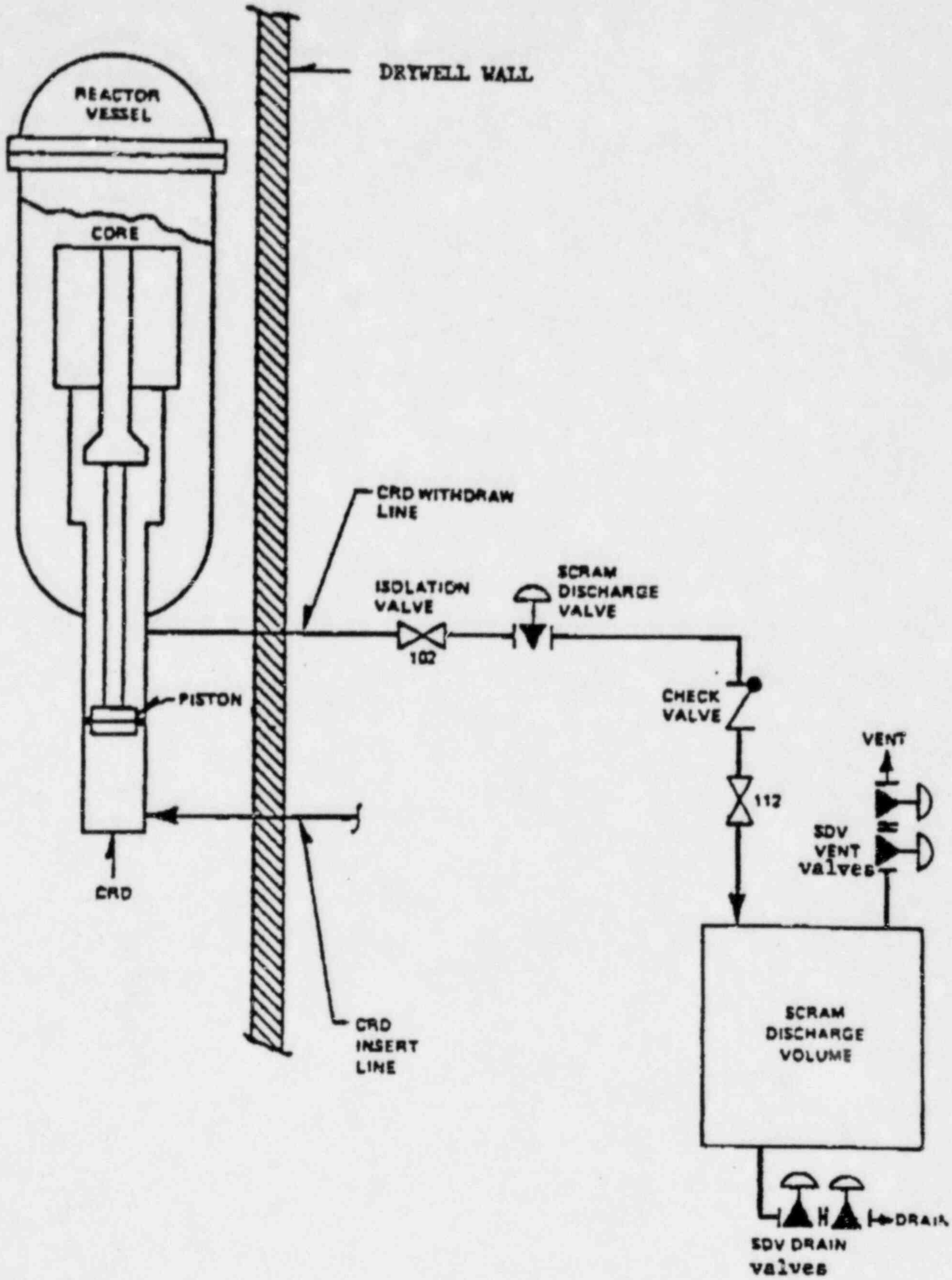

Figure 1. Simplified Schematic of Scram Discharge System

keeping with its frequency of exposure to the reactor coolant pressure boundary.

In addition to meeting the requirements of Section III of the ASME Code, the entire scram discharge system at PNPP will be subjected to periodic tests and inspections in accordance with the requirements of Section XI of the ASME Code. These examinations and tests provide additional assurance that the integrity of the system is maintained throughout the operational lifetime of the plant.

Furthermore, experience confirms the quality level and integrity of the scram discharge system piping. In more than 390 reactor-years of experience, including some with over 20 years of reactor operation, there have been no reported incidences of scram discharge system pipe cracks, leaks or ruptures. Given that the PNPP scram discharge system is designed to the stringent requirements of Section III of the ASME Code, which are equal to or surpass the requirements imposed on current operating plant scram discharge systems, similar failure free performance is expected at PNPP.

In the highly unlikely event that an SDV rupture occurred following a scram, the plant operator would be made aware of the break by safety grade plant instrumentation and would take appropriate actions to terminate the event. Although such operator actions would expedite the termination of the event, no specific operator actions are required since the potential consequences of an SDV rupture are bounded by the ECCS and containment LOCA evaluation design bases.

Sensors in the Containment Atmosphere Monitoring System and the Area Radiation Monitoring System will provide indication of any significant SDV leakage. The Containment Atmosphere Monitoring System contains the following sensors that will provide indications of SDV leakage: temperature sensors located in the same quadrant of the containment as the SDV's and HCU's approximately 25 feet above and 7 feet below the elevation of the SDV's; and, a moisture sensor 70 feet above the

SDV elevation. The Area Radiation Monitoring System provides two area radiation detectors which are located in the same quadrant and at the same elevation as the SDV's and HCU's.

Detectors also are provided for containment pressure, containment atmosphere activity, and containment purge vent activity. Increases in these parameters also will provide indication of any significant SDV leakage.

Once the plant operator recognizes the potential existence of an SDV break, he would attempt to isolate the SDV by resetting the scram (i.e., if scram reset had not already been accomplished in accordance with normal operating procedures). Resetting the scram will return the CRD system to its normal operational configuration which includes the automatic closure of the individual scram discharge valves to isolate the SDV from the reactor pressure vessel. This would terminate the leak. The operator would confirm proper scram reset by reviewing the scram valve position indicator lights on the main control console and confirming that all scram valves are shut. If the scram reset could not be accomplished because of some off-normal situation, the operator would take actions to depressurize the RPV and close the manual isolation valves on the individual hydraulic control units (HCU).

Even if the operator did not recognize the SDV pipe break as the cause of the event, the plant operating procedures would still assure the orderly termination of this event. During a postulated SDV pipe break event the drywell pressure will increase along with the containment pressure as the result of flow through the drywell vacuum breaker valves. Once the drywell pressure reaches the high pressure trip, nominally 2 psig, the plant operator would begin implementing the station emergency operating procedures which will include direction for a controlled depressurization. All equipment required to complete the depressurization and maintain water level is safety related and qualified for the LOCA environment so that the postulated SDV break will neither interfere nor affect the ability to depressurize and achieve an orderly plant shutdown. Following vessel depressurization, maintenance crews could enter and terminate the event through closure of the HCU isolation valves, if necessary. Since there are no long-term safety consequences associated with continued discharge from the postulated SDV pipe break, there is no particular time by which these actions must be taken to isolate the break.

The potential consequences of a postulated SDV pipe break are within the PNPP primary containment and emergency system design bases. The SDV at PNPP is located inside the primary containment, as shown in Figure 2. Water discharged from an SDV pipe break will cascade down, through the open grating of the HCU floor, into the suppression pool below and will be confined within the primary containment and available for ECCS suction. Similarly, steam and airborne radionuclides potentially discharged from an SDV pipe break would be confined within the primary containment which would be isolated manually by the operator or automatically from a high containment ventilation radiation signal. If required, operation of the redundant, safety grade systems will initiate containment sprays to maintain containment pressure and temperature within design limits. All normal make-up systems and emergency core cooling systems (ECCS) are located outside the primary containment in the auxiliary building, as shown in Figure 3. Therefore, the postulated SDV pipe break presents no unique challenge to the ECCS and the long-term operability of ECCS for core cooling is assured.

The radiation levels inside the primary containment will vary depending upon the size of the SDV pipe rupture, the initial concentration of radionuclides in the reactor coolant water and the rate at which the vessel is depressurized. In any case, the levels will be small compared to those employed in the PNPP containment design basis LOCA evaluations. Since this event presents no unique challenges to the ability to depressurize the vessel, maintain reactor water level and effect an orderly shutdown, there is no need for immediate isolation of the break so that manual isolation of the

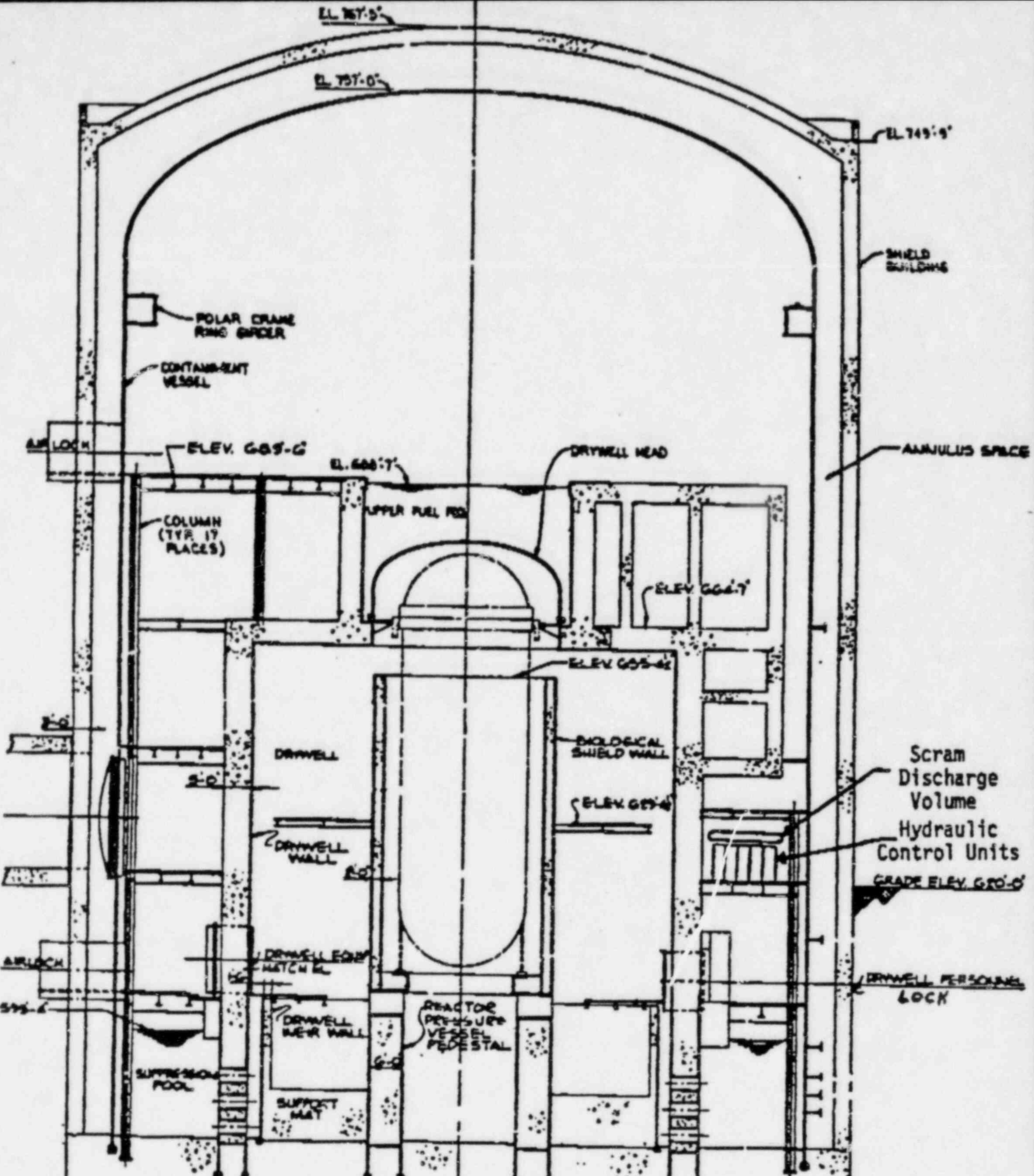

FIGURE 2. REACTOR BUILDING SECTION

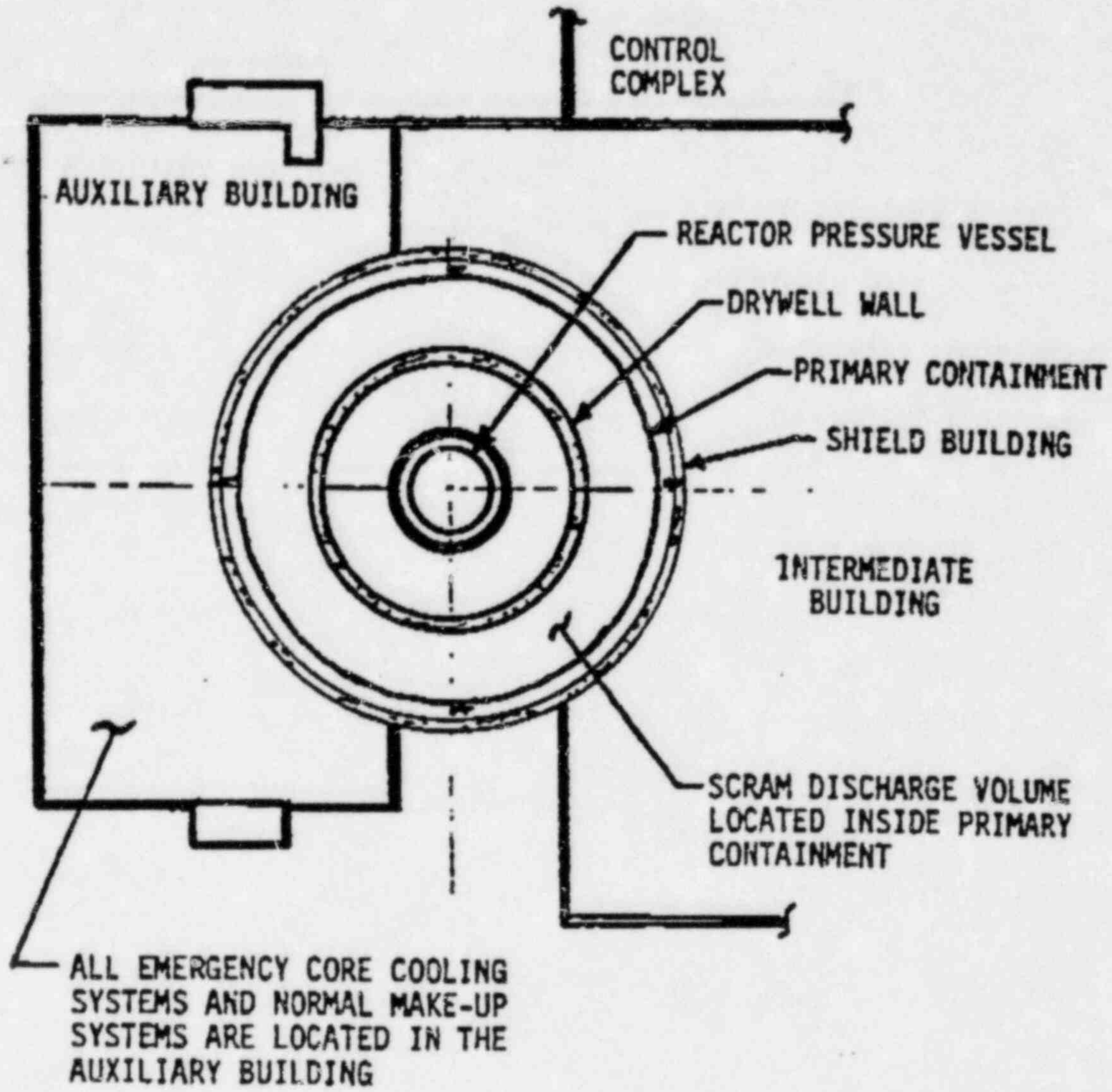

FIGURE 3. GENERAL ARRANGEMENT PLAN VIEW

SDV pipe break can be accomplished whenever the dose rates have decayed sufficiently to permit personnel entry into the containment.

To further demonstrate the adequacy of the SDV design at PNPP and to further establish that no special provisions need be made to prevent or mitigate a postulated SDV pipe break, the PNPP design was evaluated against the relevant NRC recommendations summarized in Table 5.1 of NUREG 0803. These evaluations confirm PNPP compliance to all applicable recommendations as shown in Table 1.

TABLE 1

NRC Recommendation	PNPP Compliance
(1) Period inservice inspection and surveillance for the SDV system	(1) ISI program per requirements of 1980 (or later) Edition of Section XI ASME Code
(2) Threaded joint integrity	(2) No threaded joints in SDV design
(3) Seismic design verification	(3) Entire SDV is Seismic Category 1
(4) HCU-SDV equipment procedures review	(4) Not mandatory since SDV is located inside primary containment
(5) Environmental qualification of prompt depressurization function	(5) Have safety grade ADS including environment qualification of all equipment
(6) As-built inspection of SDV piping and supports	(6) Conducted as part of established Quality Assurance Program
(7) Improvement of procedures	(7) Covered by commitment to develop procedures consistent with Owner's Group EPGs
(8) Verification of equipment design for water impingement	(8) Not required - no make-up system or depressurization system equipment vulnerable to water impingement or wetdown from SDV pipe break
(9) Verification of equipment qualified for wetdown of 212°F water	(9) See response to 8
(10) Verification of feedwater and condensate system operation independent of the reactor building environment	(10) Not required - all ECCS and normal make-up systems are located outside primary containment whereas the SDV is located inside primary containment

NRC Recommendation	PNPP Compliance
(11) Evaluation of availability of HPCI-RCIC turbines due to high ambient temperature trips	(11) Not applicable - At PNPP all ECCS and normal make-up systems are located outside the primary containment whereas the SDV is located inside primary containment - the postulated SDV pipe break presents no unique challenge to the availability of these systems.
(12) Verification of essential components qualified for service at 212°F and 100% humidity	(12) Essential components within primary containment required for orderly shutdown and termination of SDV pipe break event are qualified to bounding containment design basis environments
(13) Limitation of coolant iodine concentration to Standard Technical Specification values	(13) PNPP operating technical specifications to be developed based upon Standard Technical Specifications

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

Monty A. Ross, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 1st day of December, 1982.

Monty A Ross

Monty A. Ross
General Electric Company

Subcribed and sworn before me this 1st day of December, 1982.

OFFICIAL SEAL
KAREN S. VOGELHUBER
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My Commission Expires Dec. 21, 1984

Karen S. Vogelhuber

NOTARY PUBLIC, STATE OF CALIFORNIA

Exhibit A

EDUCATION AND PROFESSIONAL QUALIFICATIONS

Monty A. Ross

Mr. Ross is a manager in the Plant System Design organization of the General Electric Nuclear Energy Business Group, in San Jose, California. His employment with General Electric began in 1972, as an Engineer in the Design Engineering section, where he worked on the design and analyses of pressure vessel components, nuclear piping systems, refueling and servicing tools.

Starting in 1975, Mr. Ross participated in a career developing program of rotating assignments. Major activities while on this program included the experimental testing of primary containment designs in the evaluation of the thermodynamic transients which may (hypothetically) occur within the primary containment as a result of a LOCA and non-LOCA events.

In February 1979, he took the position of Lead System Engineer (LSE) for the Rod Control System. As the LSE, he was responsible for the design definition of the Rod Control System. Major tasks in this position included gaining NRC acceptance of the Control Rod Drive System return line removal and directing the evaluation and design changes resulting from the Browns Ferry 3 partial scram insertion of June 28, 1980.

In October 1980, Mr. Ross assumed a management position in the system design organization. The group that he managed, through July 1982, was responsible for the design definition of six (6) BWR Standard Plant systems including the Rod Control System. In that position, he was the lead technical contributor in the evaluation of the NUREG-0785 concerns regarding pipe breaks in the BWR Scram Discharge Piping.

Mr. Ross is a 1972 graduate of the University of California at Davis, with a BS Degree in Mechanical Engineering (power generation option) and in Material Science. In 1977, he received an MS Degree in Mechanical Engineering from the University of Santa Clara. Mr. Ross is a registered Professional Engineer in the State of California.

December 3, 1982

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CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing "Applicants' Answer in Support of NRC Staff's Motion for Summary Disposition of Issue No. 5", were served by deposit in the United States Mail, First Class, postage prepaid, this 3rd day of December, 1982, to all those on the attached Service List.

Jay E. Silberg

JAY E. SILBERG

DATED: December 3, 1982

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of THE CLEVELAND ELECTRIC ILLUMINATING COMPANY (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket Nos. 50-440 50-441

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