ATTACHMENT 5

INDIANA MICHIGAN ELECTRIC COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

January 15, 1980
AEP:NRC:00253D

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

The attachment to this letter provides information concerning the proposed design for reactor coolant system venting in compliance with the requirements of NUREG-0578. This letter fulfills the commitment we made in our letter of October 24, 1979 (AEP:NRC:00253) to provide you this information.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

JED:em

cc: R. C. Callen
G. Charnoff
R. S. Hunter
R. W. Jurgensen
D. V. Shaller - Bridgman

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ATTACHMENT TO

AEP:NRC:00253D

DONALD C. COOK NUCLEAR PLANT
DOCKET NOS. 50-315 & 50-316
LICENSE NOS. DPR-58 & DPR-74

NRR ITEM (2.1.9) - REACTOR COOLANT SYSTEM VENTING

Reactor Head Vent System

The reactor head vent system, shown in Figure 1, consists mainly of two redundant 1-inch vent pipes connected to the existing reactor head vent. Each vent pipe consists of a flow restricting orifice and two "fail closed" isolation valves, which during normal plant operation are normally closed. The redundant valves are powered from separate trains to assure operation when required and the "fail closed" design of the valves assures isolation. The system is designed such that any single active failure of a component will not prevent vessel gas venting nor prevent isolation of the venting path. The system is capable of being dismantled in a relatively easy manner for refueling, without breaking the RCS pressure boundary and maintains the necessary manual venting function during vessel filling operation. The valves are solenoid-operated isolation valves with Class IE operators qualified to IEEE-323 (1974).

The system connects to the reactor vessel head via the existing vent pipe with redundant flow paths through 3/8 inch orifices. The 3/8 inch orifice restricts the flow rate from a pipe break downstream of the orifice to the makeup capacity of one charging pump. The reactor vessel head vent system isolation valves will be supported from the seismic support platform and can be disconnected downstream of the second isolation valve to accommodate refueling. In this manner, the necessary flanged connections will be outside the reactor coolant pressure boundary. All piping and valves upstream of the flanges will remain integral with the reactor vessel head at all times.

The reactor head vent will discharge into an area of the containment which will provide adequate dilution of any combustible gases.

Pressurizer Vent System

The pressurizer vent system provides venting of the pressurizer using safety grade equipment. As shown in Figure 2, two redundant 1-inch vent pipes with two "fail closed" isolation valves on each, that during normal plant operation, will be kept normally closed. The redundant valves are powered from spearate trains to assure operation when required; the "fail closed" design of the valves assures isolation. The system is designed such that any single active failure of a component will not prevent pressurizer venting nor prevent venting isolation.

The system connects to an existing 3/4 inch vent line on the pressurizer safety valve piping, which is normally filled with steam. An orifice, if required, is being considered to limit the blowdown from a pipe break to the makeup capacity of one charging pump.

The valves are solenoid operated isolation valves with Class IE operators, qualified for operability to IEEE-323 (1974).

The discharge of the pressurizer vent system will discharge into an area of the containment which will provide adequate dilution of any combustible gases.

Piping Analysis

The piping analysis will incorporate a three dimensional dynamic model which will include the effects of interaction with the reactor coolant system and the effect of the supports and supported equipment. The piping analysis of the static and dynamic model will employ the displacement method, lumped parameter, stiffness matrix formulation and assumes that all components and piping behave in a linear elastic manner.

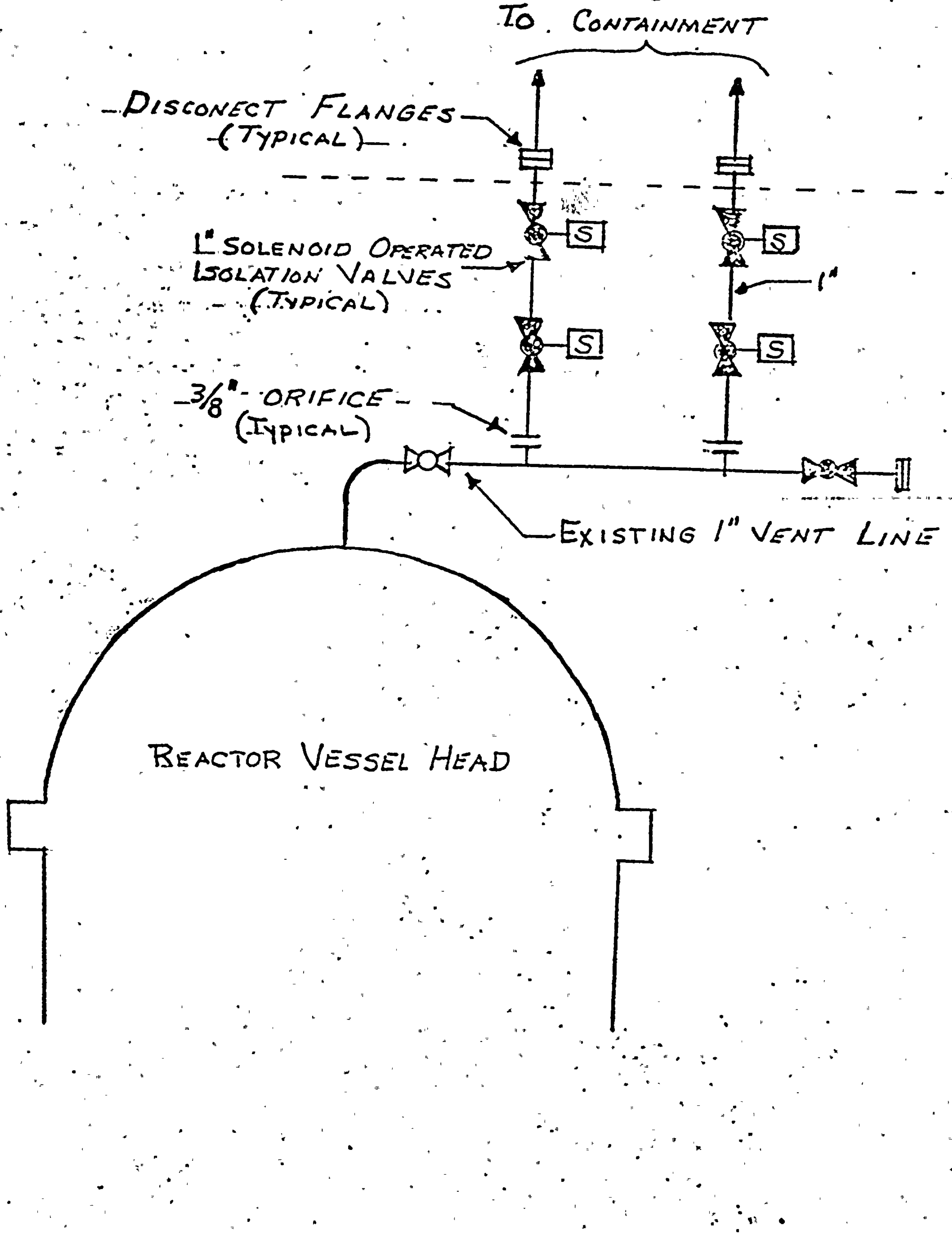


FIGURE 1

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

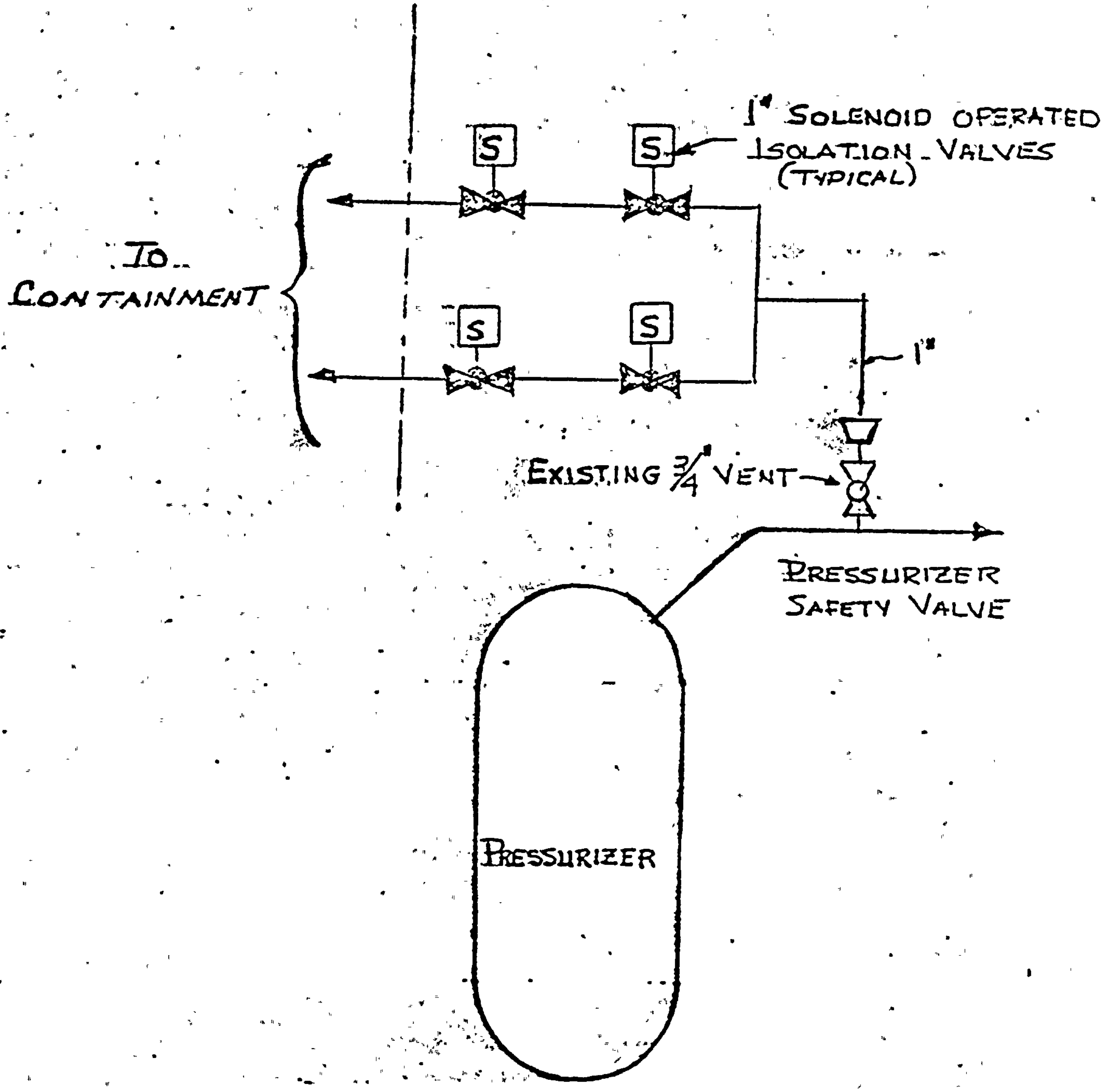


FIGURE 2

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

ATTACHMENT 6

TO

AEP:NRC:00334

ATTACHMENT 6

This attachment provides information in addition to that previously submitted.

ITEM 2.1.2

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Force, submitted to the NRC a document entitled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems, December 13, 1979." We consider the program to be consistent with the requirements of NUREG-0578, as supplemented. Indiana & Michigan Electric Company (I&MECo.) is participating in this program and will follow its results.

ITEM 2.1.3.b

The Westinghouse Owners' Group, of which I&MECo. is a member, has performed analyses (as required by Item 2.1.9) to study the effects of inadequate core cooling. These analyses were provided to the NRC "Bulletins and Orders Task Force" for review on October 31, 1979. As part of the submittal made by the Owners' Group, an "Instruction to Restore Core Cooling during a small LOCA" was included. This instruction provides the basis for procedure changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. We have incorporated the key considerations of this instruction into our LOCA procedures and have provided training to the operators in this area.

The submittal referenced above described the capabilities of the core exit thermocouples in determining the existence of inadequate core cooling conditions and their superiority in some instances to the loop RTDs for measuring true core conditions.

We have incorporated, and have operational, a program in the PRODAC 250 computer system which provides information on saturation margin to the operators. Discussions held on December 20, 1979 with members of the NRC staff required that the margin to saturation be recorded by the trend typewriter every one (1) minute interval and continuously whenever conditions are such that close monitoring of subcooling is necessary. This has been implemented. These actions comply with the requirements of NUREG-0578. We do plan to substitute a subcooling meter (as described in Attachments 1, 2, 3 and 4) for the computer system. This work should be complete prior to April 1, 1980.

Reactor vessel water level is also being investigated to determine if it is practical application for indicating the approach to or the existence of inadequate core cooling. Several systems for determining water level are under review by the Westinghouse Owners' Group. The final system configuration will be evaluated to assess its usefulness in providing information to the operator for proper operation of a vessel venting system and for normal water level control.

ITEM 2.1.4

The information requested in this NUREG-0578 item has previously been provided in our letters of May 1, 1979 (AEP:NRC:00185) and July 25, 1979 (AEP:NRC:00185B). Our responses to Action Items 4, 7a, and 9 of IE Bulletin 79-06A, Revision 1, contained in the above mentioned letters, shows complete compliance with the NRC positions in NUREG-0578.

ITEM 2.1.6.a

We will complete by January 31, 1980, or within 30 days after the Units are back in operation, the leak test surveillance inspection program on all the systems or portions of systems in accordance with the requirements of NUREG-0578 as supplemented by your October 30, 1979 letter. The results will be available for NRC inspection at the Plant.

ITEM 2.1.8.a

We will submit to you a preliminary description of conceptual plant modifications required for post-accident sampling by January 31, 1980.

ITEM 2.1.8.b

Existing plant spectrum analysis equipment will be used for determining airborne iodine concentrations in accordance with Mr. Denton's October 30, 1979 letter. No new equipment is being procured specifically for this purpose.

ITEM 2.1.9

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basis by the Westinghouse Owners' Group. of which I&MECo. is a member. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners' Group on June 29, 1979. Incorporated in that report were guidelines that were developed as a result of small break analyses. These guidelines have been reviewed and approved by the B&O Task Force and have been presented to the Owners' Group utility representatives, including I&MECo., in a seminar held on October 16 - 19, 1979. Following this seminar, we have developed plant specific procedures and trained personnel on the new procedures.

The work required to address the other two areas (inadequate core cooling and other transient and accident scenarios) has been performed in conjunction with schedules and requirements established by the Bulletins and Orders Task Force. Analysis related to the definition of inadequate core cooling based on exiting plant instrumentation and for restoring core cooling following a small break LOCA was submitted on October 31, 1979. Work in this area is continuing.

With respect to other transient and accidents contained in Chapter 14 of the D. C. Cook FSAR, the Westinghouse Owners' Group has performed an evaluation of the actions which occur during an event by constructing event trees for each of the non-LOCA and LOCA transients. From these event trees a list of decision points for operator action has been prepared, along with a list of information available to the operator at each decision point. Following this, criteria have been set for credible misoperation, and time available for operator decisions has been qualitatively assessed. The information developed was then used to test Abnormal and Emergency Operating Procedures against the event sequences and determine if inadequacies exist in the AOPs and EOPs. The results of this study will be provided by the Owners' Group to the Bulletins and Orders Task Force as required by them.

The Owners' Group has also provided test predictions analysis of the LOFT L3-1 nuclear small break experiment. This analysis was provided on December 15, 1979, in accordance with the schedule established mutually with the Bulletins and Orders Task Force.

ATTACHMENT 7

TO

AEP:NRC:00334

ATTACHMENT 7

ITEM 2.1.8.a

Post - Accident Sampling

In accordance with the discussion held on December 10, 1979 with members of the NRC Staff, we provide the following information concerning the minor plant modifications performed to allow for post-accident sampling.

Reactor Coolant Liquid Sampling

For each unit, there is a common sample from either the Loop No. 1 or 3 hotleg of the reactor coolant system to a failed fuel detection system. Modifications have been made for post-accident sampling by teeing into the failed-fuel-detection sample line downstream of the sample heat exchanger outlet (see Figure 1). The sample tap is run to a shielded sample collection area. The sample will be collected in a lead pig and transported for analysis. A demineralized water line is provided for flushing after sample collection.

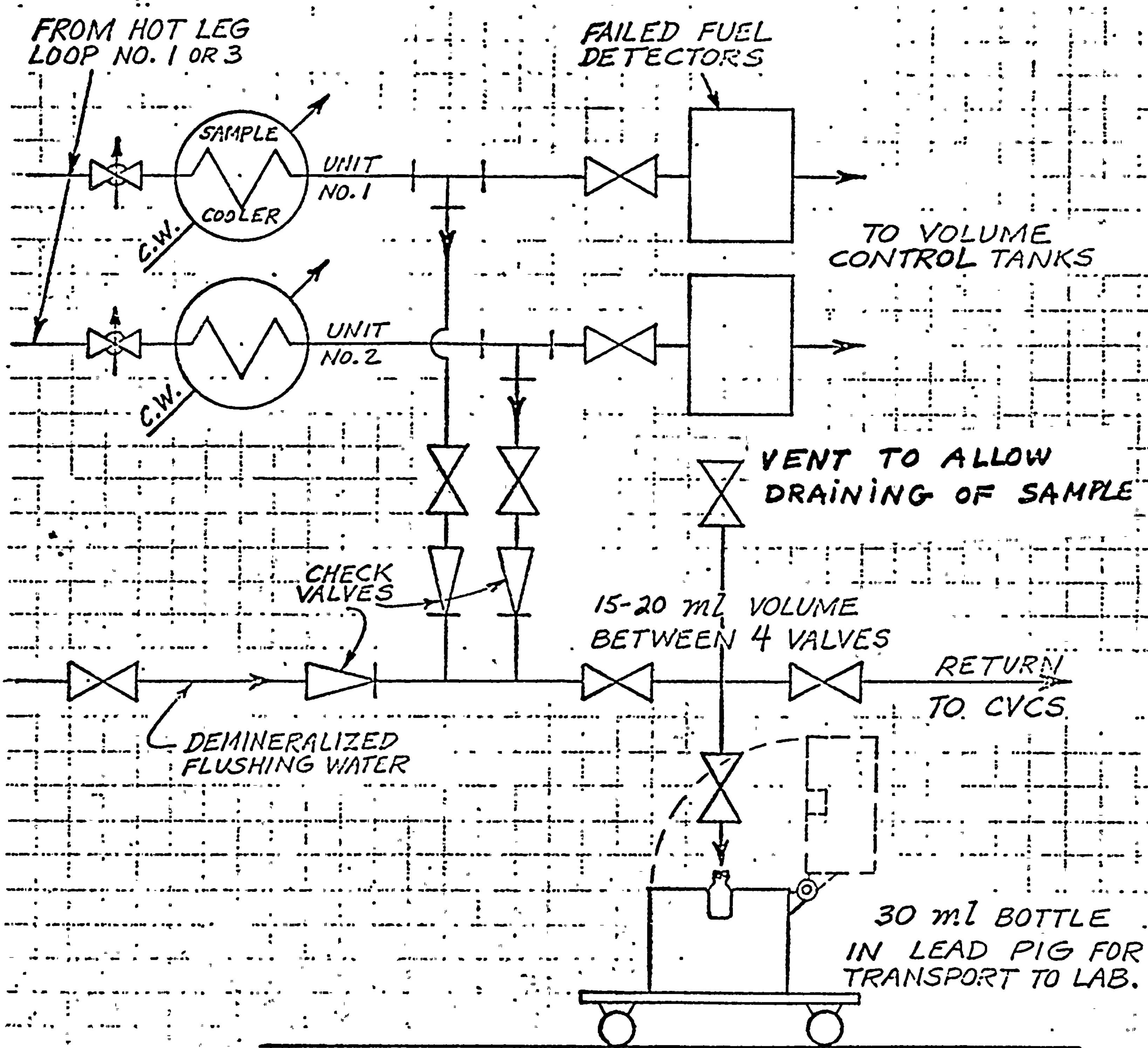
Containment Atmosphere Sampling

A sample from the containment atmosphere is supplied to the air particle and radiogas detectors. To provide for post accident sampling of containment atmosphere, a line has been "teed" into the air particle detector sample line downstream of the containment isolation valves (see Figure 2). The sample is pumped to a shielded sample collection area. The sample will be collected in a lead pig which will have provisions for housing a noble gas/hydrogen sampling vessel. The sample will be transported elsewhere for analysis. A purge gas line is provided for purging the cartridge of noble gases.

DONALD C. COOK NUCLEAR PLANT UNITS 1 & 2

FIGURE 1

SUBJECT POST ACCIDENT SAMPLING-REACTOR COOLANT LIQUID



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SUBJECT POST ACCIDENT SAMPLING - CONTAINMENT ATMOSPHE[RE]