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October 8, 1984

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

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

On September 14, 1984 representatives from Duke Power Company, Westinghouse Electric Corporation and the NRC Staff met at the NRC's office in Bethesda, Maryland to discuss Proposed License Conditions 14a and 17 which concern the main steam line break (MSLB) inside containment. The purpose of this letter is to provide a summary of the justifications for interim operation of Catawba Unit 1 pending completion of the ice condenser drain test and the multi-node containment analysis which are being performed by Westinghouse to support the containment analysis methodology and are described in Attachments 2 and 3.

For a MSLB inside containment, Westinghouse has revised its methodology used to calculate the mass and energy released into the containment to consider the calculation of superheated steam following tube bundle uncovery. Modifications were also made to the Westinghouse methodology used to calculate the containment temperature response. These modifications were described in a May 18, 1984 submittal from Westinghouse to the NRC on WCAP-8354-P, Supplement 2, "Westinghouse Long Term Ice Condenser Containment Code-LOTIC-3". After discussions with the NRC Containment Systems Branch (CSB) staff, several modifications were made to the revised model and were described in a September 10, 1984 submittal from Westinghouse to the NRC. Further modifications, which address additional NRC/CSB staff concerns, are outlined in Attachment 1. With these revisions, the calculated peak containment temperature is 324°F, which is below the original FSAR peak containment temperature.

Westinghouse has also performed a preliminary fracture mechanics evaluation to determine a maximum crack opening area for a MSLB. This evaluation concluded that tube bundle uncovery would not occur and therefore the original equipment qualification temperatures envelopes would not be exceeded. This evaluation is described in Attachment 4.

DWNGS TO: K. JABBOUR

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As requested in the September 14, 1984 meeting, a review of the critical pieces of equipment located in the lower containment has been performed and is included as Attachment 5. This analysis demonstrates that even if the qualification temperature of critical equipment were exceeded, required safety functions could still be accomplished.

Based on the justifications noted above, and a revised containment analysis which has shown that expected temperatures in the lower containment as a result of a MSLB would not exceed the current FSAR analysis nor the envelopes used for equipment qualification, it is concluded that plant safety would not be adversely affected in the event of a MSLB inside containment. It is requested that Proposed License Conditions 14a and 17 be revised to require submittal of a revised MSLB analysis prior to the first refueling outage.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS:slb

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
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Attachment 1
Catawba Nuclear Station

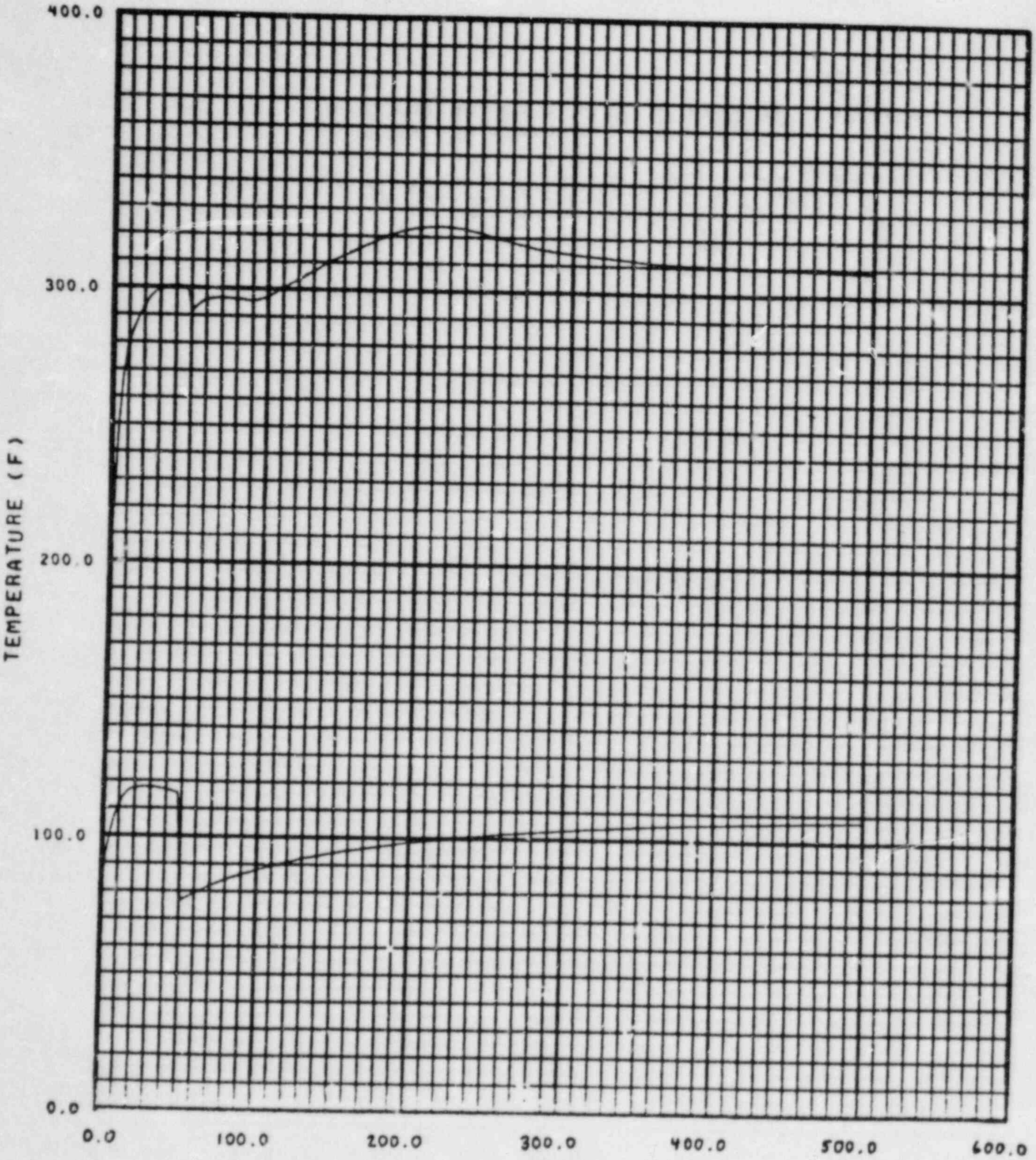
MODI . .TION TO THE CONTAINMENT ANALYSIS METHODOLOGY

In a Westinghouse submittal to the NRC, dated 9/10/84, several modifications to the approved containment methodology were described. In a meeting between the NRC/CSB staff and Westinghouse, the staff expressed a concern over the modifications made to the wall heat transfer coefficients. To address that concern, Westinghouse set the wall condensing heat transfer coefficient to a maximum value of 280 $\frac{BTU}{hr\text{-}ft^2 \cdot {}^\circ F}$ which is the maximum value of the Uchida correlation, which is acceptable to the staff. Westinghouse also changed the wall convective heat transfer coefficient from a value of 2 BTU/hr-$ft^2 \cdot {}^\circ F$ to the natural convection heat transfer correlation described in the CONTEMPT4/MOD^2 report, TREE-NUREG-1202, dated February 1978.

The ice condenser drain model described in the Westinghouse submittal to the NRC dated 5/10/84 was also changed slightly to reflect some additional test data obtained since 5/10/84. The changes basically increased the width of the film on the reactor coolant pumps and the steam generators. This model will be verified by the test plan described in Section 4 of this report.

With these modifications the Catawba Unit 1 lower compartment temperature transient is given in Figure 1.1. The peak temperature is 324°F. A sensitivity study was also done to determine the impact of the latest drain modifications. A case was run with the lower heat transfer coefficients described above and the drain model described at the Duke-Westinghouse-NRC-CSB meeting held on 9/14/84. This case produced a peak temperature of 327.0°F, which is the same as the original FSAR.

Fig. 1.1



COMPARTMENT TEMPERATURE

Attachment 2
Catawba Nuclear Station

PROPOSED MULTINODE CONTAINMENT ANALYSIS

A multinode analysis of an ice-condenser containment will be performed to determine if temperature gradients develop in the lower compartment following a main steam line break as a result of the azimuthal location of the break. The analysis should model

1) passive heat sinks

2) source point for mass and energy release into the containment

3) the condensation and resulting ice melt at different aximuthal locations within the containment

4) the resulting drain flow from the ice melt and heat transfer to this drain flow using the results of the drain test as a basis for a drain model

5) the noncondensable gas distribution within the containment

6) the resulting containment atmosphere temperature as a function of time.

The COBRA-NC computer code will be used to perform this analysis. This code was developed for containment analysis and it has been assessed against a variety of containment experiments. These comparisons have demonstrated the capability of the code to calculate the condensation, pressurization and noncondensable gas distribution within the containment. COBRA-NC is a two-component, two-fluid, three-field representation of two-phase flow. The two-components are water with its vapor and a mixture of non-condensable gases. The two-fluids are liquid water and the mixture of steam and non-condensable gases. The three velocity fields are those for the vapor/gas mixture, the liquid film and suspended liquid drops. The gas mixture may be composed of any number of non-condensable gases including air. The code is formulated in three-dimensional Cartesian coordinates. Three temperature fields are solved in the code; one for the vapor/gas mixture, one for the combined liquid phases and one for solid structures.

Both the hydrodynamic and heat transfer equations are solved using a finite difference solution scheme on an Eulerian Mesh. The conservation equations for mass, momentum and energy are solved using a semi-implicit Block-Newton-Raphson method. The conduction in solid structure is solved by direct inversion. Turbulence is modeled using the mixing length theory and is similar to COBRA/TF or COBRA-TRAC.

A general heat transfer correlation selection logic is available in the code. The Uchida correlation is used for the condensation heat transfer coefficient to dry structures. A heat transfer coefficient of:

$$h_l = \frac{2k_l}{\delta}$$

is used for wetted walls where k_l is the thermal conductivity of the liquid film and δ is the thickness of the liquid film.

The maximum of the Uchida correlation:

$$h = 79.53 \left(\frac{\rho_v}{\rho_g}\right)^{0.8}$$

and the film/vapor interfacial heat transfer coefficient is then used to calculate the interfacial heat transfer between the containment atmosphere and the liquid film. This treatment of the condensation heat transfer avoids any double accounting for structures that are covered with water.

The code has the capabilities that are required to perform the analysis. The code may need to be slightly modified so that the drain flow drop size and velocity can be specified from the drain test. The code normally calculates these values internally based on local fluid conditions. The models or changes which are currently planned to be made to COBRA-NC include:

- use of the calculated mass and energy release from the W steam line break analysis

- use of the ice bed model on LOTIC-III and the Waltz Mill tests

- use of the LOTIC-III drain model and drain data from the Forest Hills tests.

The exact representation of the drains in COBRA-NC cannot be made at this time since additional tests with Steam Generator Reactor Coolant Pump, and cable tray simulations should be completed.

The noding that we plan to use for the analysis is shown in Figure 1. This noding provides 12 azimuthal nodes in the lower compartment. There are an additional 6 nodes in the lower compartment to allow azimuthal convection currents to be modeled. The vertical mesh is shown in Figure 2. There are four vertical nodes in the lower compartment. Thus, with four vertical levels each having 18 nodes per level, this gives a total of 72 nodes in the lower compartment. There are five vertical levels in the ice bed giving a total of 60 nodes in the ice bed.

The dome region will be modeled with 4 nodes on each level and will be 4 nodes high in the ice bed region and 3 nodes high in the dome region giving a total of 28 nodes. Other rooms surrounding these major compartments will be modeled with lumped parameter volumes. This noding should be sufficient to demonstrate if any azimuthal gradients will develop.

If severe gradients are calculated to occur, more detailed noding can be used to assess if any important instrumentation or equipment is located in the region of higher temperatures. Data and models from the full scale drain experiments will be put into the code to model the drain sheet, droplet field, and interactions with cable trays, steam generators, and reactor coolant pumps.

The results of the analysis will be compared to the single volume LOTIC-III containment model to assess gradient and local effects. We feel that the above multinode calculation will address any residual questions on mixing and non-condensible effects within the containment.

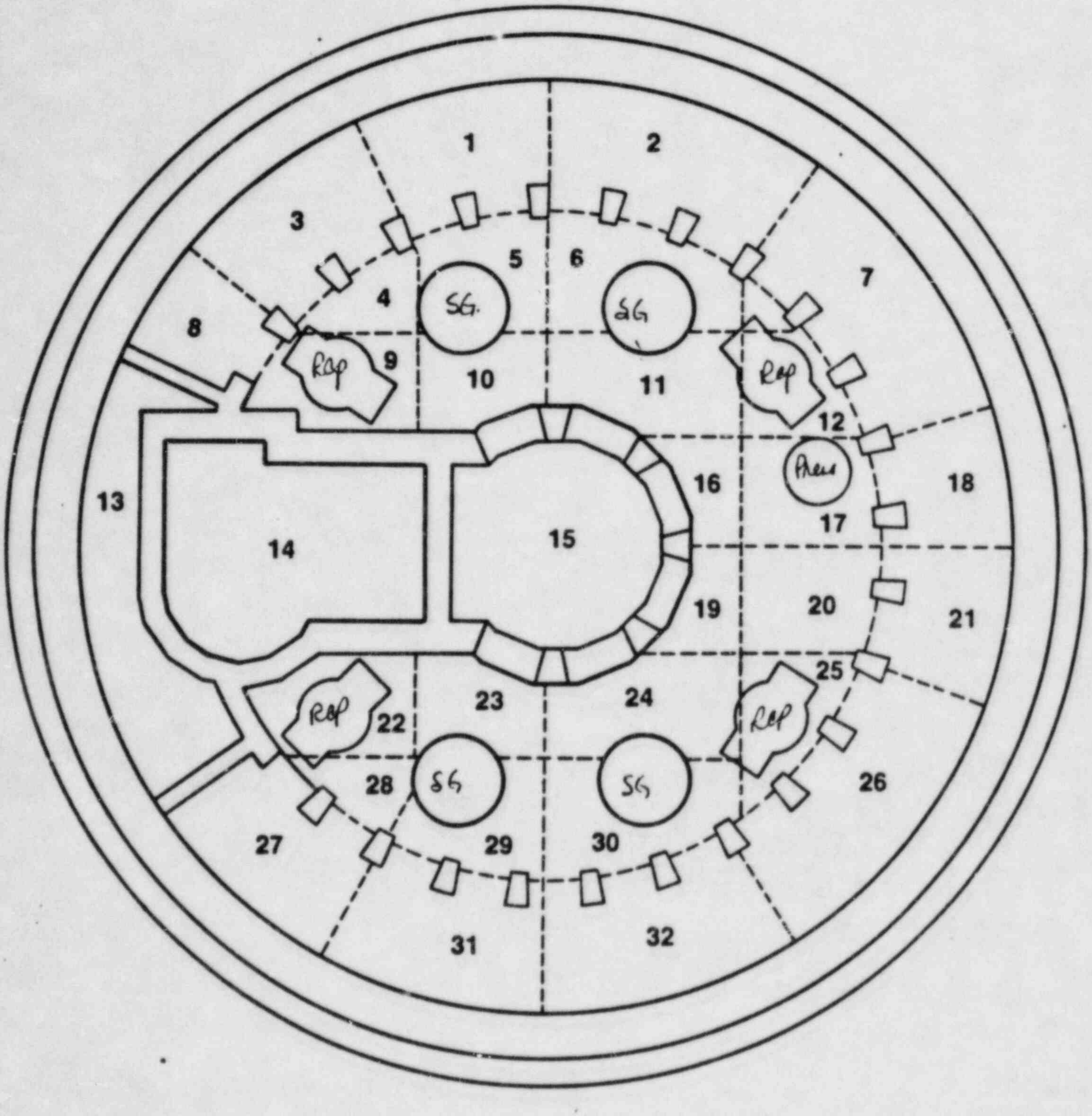


Figure 1 Noding at Elevation of Ice-Bed Inlet

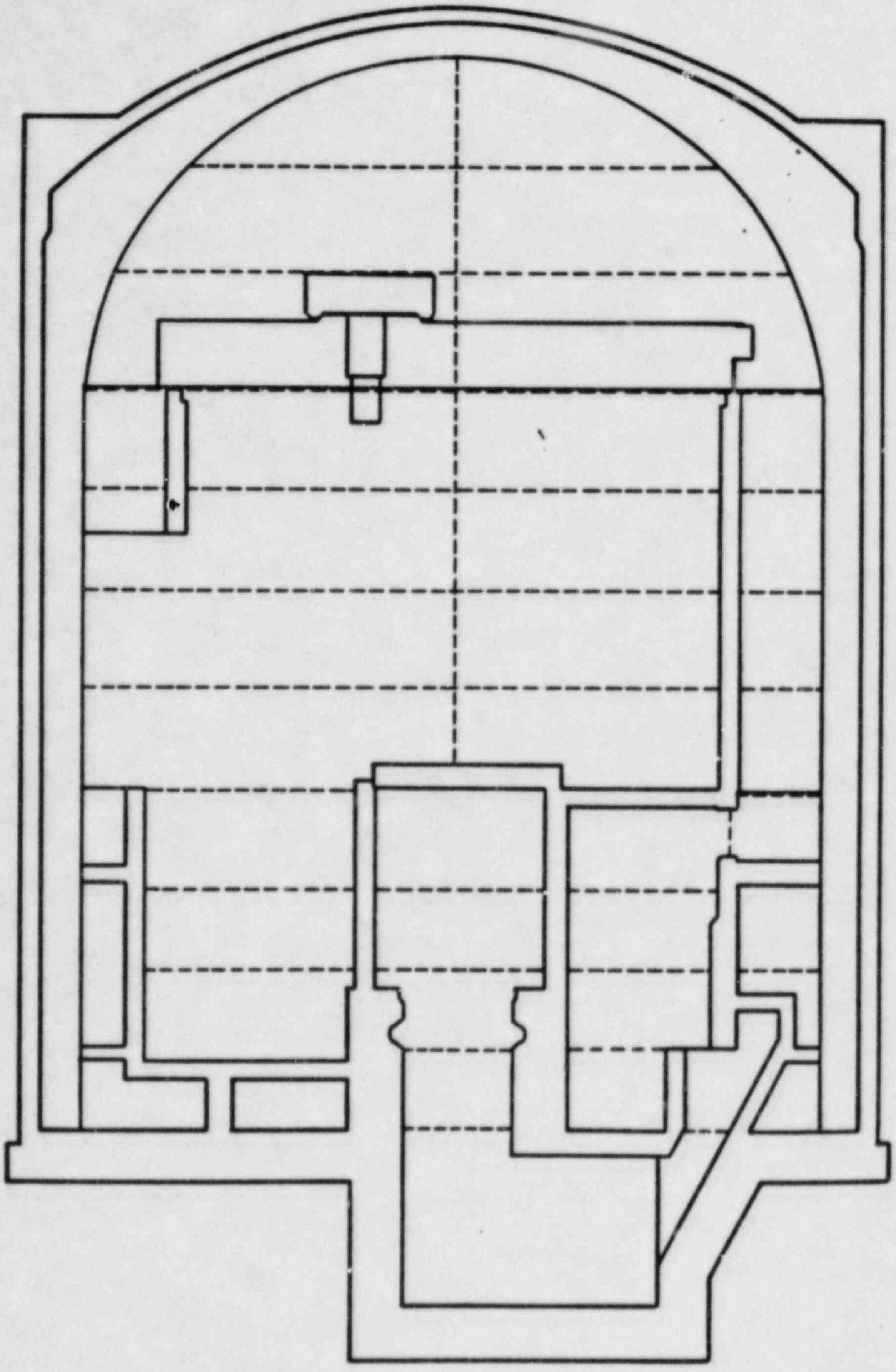


Figure 2 Vertical Noding

ICE CONDENSER DRAIN TEST PLAN

INTRODUCTION

The objective of the ice condenser drain flow tests is to provide data to characterize the hydraulic effects of the ice condenser drains such that a model of the drain flow can be verified for use in the Westinghouse LOTIC-III containment code. The test program will use an ice condenser drain valve obtained from Duke Power as well as a full scale representation of the drain piping and valve connections. All elevations and pipe lengths are representative of ice condenser drains found in the Westinghouse ice condenser plants.

DESCRIPTION OF PROPOSED TESTS

From the two trips to Catawba and discussions with the NRC, it appears that there should be five configurations tested in the ice condenser drain tests:

1) Free fall of the drain discharge through simulated cable trays with a few vertical structures;

2) Free fall of the drain discharge onto a flat surface;

3) Free fall of the drain discharge onto a reactor coolant pump simulation;

4) Drain discharge impacting a simulated steam generator shell;

5) Free fall of the drain discharge with nothing under it: these tests will act as a reference.

The present configuration for the ice condenser drain test is with the drain situated approximately 33 feet in the air on the top of one of the buildings at the Forest Hills test site. There is a continuous flow path from a pool which was constructed on the driveway of the laboratory entrance, through a pump to a large open-ended flow box on the roof of the building. The flow can be controlled to the box over the range of 1800-100 gpm. The flow empties into the box then drains through a one-foot square opening into the 12-inch inside diameter drain pipe. The flow then exits the drain pipe through the valve and falls to the pool located 33 feet below. A photograph of the facility is shown in Figure 1.

Since the drain height is not full scale (33 feet versus 40 feet to the containment sump), the parameters which have been preserved as full-scale are:

1) the exact distances below the drain and from the simulated containment crane wall;

2) the distances between the bottom of the reactor coolant pump and steam generator lower head to the containment sump floor;

3) the length of the components, the generator simulator and pump are shorter than the real components to compensate for the height difference between the real drain elevation (40 feet) and the test drain elevation (33 feet).

The tests are performed in a steady state fashion at a fixed inlet flow such that photographs and high-speed movies can be used.

The data which is obtained for each test includes:

1) still photographs to give a general characterization of the flow such as jet breakup distance, jet spread, jet penetration from the wall of the building, splashing behavior, and liquid film spreading;

2) valve angle and position as a function of time;

3) liquid level at two positions in the horizontal run of the drain pipe;

4) inlet drain flow rate, water temperature and pressure (at the pump discharge);

5) high speed photography to characterize the drop diameters, splashing effects of the drain flow hitting the equipment, and drop sizes generated from falling liquid films from simulated equipment; and

6) if possible, a collection of splashed droplets.

Each configuration will be discussed in the text which follows and the test matrix for each will be given. There will be one test for each configuration in which the flow is ramped down from an initially high value to the low value such that a video tape can be made.

1) Free Fall of a Jet Through a Set of Simulated Cable Trays

This test is comprised of a variety of parts, each part having a different amount of cables in the cable tray(s) as shown in Figure 2. For all parts of this test, the top tray will be at an elevation of 7 feet beneath the drain pipe centerline. If a second tray is used, it will be positioned 15 inches under the top tray; both of these dimensions are noted in Figure 3. The first part of this test has both cable trays filled with simulated cables (see Figure 2) 1.0 inch, 1.25 inches and 1.50 inches in diameter. The second part will have the top tray full of cables and the lower tray half-filled. For the third part of this test, the lower tray will be removed and the top (filled) tray will stand alone. The last part has the single tray half-filled with cable.

The first two parts of these tests are designed to simulate the flow of water through the trays, measuring velocity and droplet sizes. For each flow in these parts, 6 still photographs will be taken illustrating the flow pattern (width, breakup, scattering, etc.) and 2 high-speed movies will be taken, one of the flow between the cable trays and one under the lower tray. If the 600 gpm flowrate causes the discharge to miss the cable tray arrangement, it will be deleted.

The third and fourth parts of this test are designed to observe and measure the splashing of the flow from the top tray. A collection device is to be built onto the cable tray structure such that the volumetric flow rate of the splashed jet can be measured. This device (e.g., canvas sheets) is to be positioned such that the splash is captured but the flow that misses the tray is not. For each flowrate indicated in Table 1, 6 still photographs and a high-speed movie will be taken and the splashed volume measured. It should be noted that this collection method may not prove workable.

The Computer Data Aquisition System (CDAS) will run through all parts of this test. The test conditions are given in Table 1, reflect a reduction in the maximum flow, and have a spacing similar to the shakedown tests. The still photographs and high-speed movies for splashing will have to be taken at the top tray elevation. The high-speed movies for parts 1 and 2 are planned to be taken at 5 feet beneath the lower cable tray after the jet has broken up.

The last test will vary the flow from 1200 gpm down to 200 gpm in steps with intermediate values of 800, 600, and 400 gpm. The sequence will be video taped with voice commentary.

2) Free Fall of the Jet Onto a Flat Surface

This test will use the test configuration of the first test series although plywood will be nailed onto the top of the cable tray so as to create a flat surface. The plywood is to have a thin metal sheet on top of it simulating flat surfaces in the lower containment. The purpose of this test series is to characterize the splashing of the discharge as it impacts the surface. Still photographs will be taken of the flow pattern as will high-speed movies at each flow rate indicated on Table 2.

3) Free Fall of the Drain Jet onto the Reactor Coolant Pump (RCP) Motor

For this test series, two types of tests should be planned. First, experiments to determine the spread of the liquid film running down the side of the simulated RCP motor. Second, repeat the tests but with the objective of characterizing the splashing nature of the jet flow from the drain valve onto the top of the RCP motor. For each of these tests, a model of a RCP with its motor will be constructed and located 3.5 feet below the centerline of the drain valve. The outside edge of the simulated RCP motor will be 5 feet from the building wall which simulates the containment crane wall. The model of the RCP motor will be a right circular cylinder 6.5 feet in diameter and 14.5 feet in height. The bottom of the simulation will be approximately 15 feet above the pool surface. The model of the RCP motor can be made of wood which should have a thin metal sheet on the surface to simulate the thermal insulation on the RCP. Once the base cases are conducted and the behavior of the flow is observed, attempts will be made to collect and measure the liquid splashout from the RCP. The current idea is to use plastic sheeting that will collect the splashed drops which then drain into a bucket in which the collection flow is timed with a stopwatch. Again it should be noted that this collection method may prove unworkable.

The test conditions for the tests in this series are given in Table 3. Again, two types of tests will be conducted. The tests to determine the film flow behavior will be done first and the tests to examine the splashing and the break up of any liquid film that forms on the simulated RCP will be conducted second. Still photographs and high speed movies around the pump motor simulation will be taken to show the splashing and film formation. High-speed movies will be taken below the simulation to measure the resulting droplet spray that will form as the liquid film falls off the simulated RCP motor.

4) Drain Flow Impacting on Steam Generators and Pressurizer

This test series is similar to series number 3 excepting that the drain flow will now impact on a solid object that blocks part or all of the flow path for the drain discharge. A steam generator simulation will be constructed as a right circular cylinder approximately 20 feet long and 12.5 feet in diameter. The simulated generator will be located 5 feet from the West wall and the top of the generator will be 2 feet above the valve top. The majority of the tests will be conducted with the steam generator on the centerline of the valve, tests labeled as "TBD" will be conducted with the generator simulation offset from the valve centerline by 5 feet (at the same height).

The first series of tests will evaluate the behavior of the liquid film flow that will run down the side of the steam generator simulation while the second series of tests will examine the splashing characteristics of the drain flow similar to test series 3. In this case, the splashed flow will be measured using the same approach as in series 3. Again, wood is the preferred material provided that the surface consists of a thin metal sheet. Testing of the SG simulation necessitates the construction of slightly more than a semi-cylindrical surface as shown in Figure 4. The test conditions for this series are given in Table 4.

5) Free Fall Tests

This last drain test will be configured as in the shakedown test with nothing below it. The tests will be steady state with the flows given in Table 5. Still photographs will be taken to obtain the jet width, breakup point into smaller liquid sheets, or ligaments, and the breakup point from ligaments into a drop field, and the jet arc or distance from the wall that the flow hits the pool. Still photographs will also be taken of the flow directly at the valve opening (side views) and of the flow in the tank. High-speed movies will be taken at the center of the free jet, five feet below the drain bottom and five feet above the pool at the jet center. The CDAS will be on and the flow, ΔP data, valve angle, and temperature data will be recorded.

DATA REDUCTION AND ANALYSIS

The exact location for the high speed movies cannot be given at this time because it is not known what the splashed and film flow looks like on each component. There will be a shakedown test to examine the flow behavior, then, based on visual observation, the high-speed photography and still photos will be assigned to different locations to best characterize the film and splashed flow. Scales or reference dimensions will be photographed for both still photos and the high-speed photography such that quantitative data on jet spread, width penetration, and breakup can be obtained. The high-speed movies will also have a reference dimension in the view such that droplets can be scaled from the reference to obtain drop size, distribution, and velocity. The current thinking is to use a shielded camera with a roof over it, use back lighting, and have a square slot cut in the roof to let a representative sample of the flow to enter and pass in front of the camera. In this fashion, the depth of field is controlled and more accurate resolution of the drops and drop distributions should be possible.

The approach used to obtain the drop sizes and distributions will be to view the high-speed film, select portions of the film to be enlarged into 8-1/2 x 11 glossy prints. These prints, with the drops and a reference dimension, will be analyzed using a Leitz-T.A.S. Texture Analyzing System. This image analysis method will yield a droplet frequency spectrum which can then be used to calculate droplet number diameters and a droplet Sauter mean diameter size. This droplet size will then be compared to those predicted by the proposed drain model in the LOTIC-III code. A similar procedure will also be used for the splashed drops, however, the lighting will be more difficult. It is felt that with both the still and high speed movie data, the drain model can be verified for the LOTIC code.

TABLE 1

CABLE TRAY/DATA REQUIREMENTS

Flow	Number of High-Speed Movies	Minimum Number of Still Photos	CDAS on	Video Tape	Comment
200	2	6	yes	no	part 1
400	2	6	yes	no	
600	2	6	yes	no	if needed
1200 - 200	-	-	yes	yes	
200	2	6	yes	no	part 2
400	2	6	yes	no	
600	2	6	yes	no	if needed
200	1 (splashing)	6	yes	no	part 3
400	1 (splashing)	6	yes	no	
600	1 (splashing)	6	yes	no	if needed
200	1 (splashing)	6	yes	no	part 4
400	1 (splashing)	6	yes	no	
600	1 (splashing)	6	yes	no	if needed
1200 - 200	-	-	yes	yes	

TABLE 2

FREE JET ONTO A FLAT SURFACE

Flow	Number of High-Speed Movies	Minimum Number of Still Photos	CDAS on	Video Tape
200	2	6	yes	no
400	2	6	yes	no
600	2	6	yes	no
1200 - 200	-	-	yes	yes

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TABLE 3

DRAIN FLOW FALLING ON A RCP MOTOR

Flow	Number of High-Speed Movies	Minimum Number of Still Photos	CDAS on	Video Tape	Comment
200	4	10	yes	no	Tests to examine the droplet flow from the RCP motor
400	4	10	yes	no	
800	4	10	yes	no	
1200	4	10	yes	no	
TBD	4	10	yes	no	
1200 - 200	-	-	yes	yes	
200	-	2	yes	no	Tests with collection scheme to get the splashed flow from the RCP
400	-	2	yes	no	
800	-	2	yes	no	
1200	-	2	yes	no	
TBD	-	2	yes	no	

TABLE 4

STEAM GENERATOR/DRAIN IMPACT TESTS

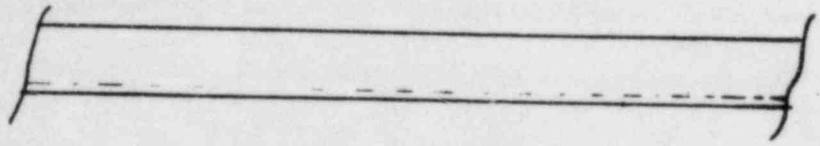
Flow	Number of High-Speed Movies	Minimum Number of Still Photos	CDAS on	Video Tape	Comment
200	4	10	yes	no	Tests to examine the droplet and splashed flow
400	4	10	yes	no	
600	4	10	yes	no	
1200	4	10	yes	no	
1200 - 200	-	-	yes	yes	
TBD	4	10	yes	no	
TBD	4	10	yes	no	
200	-	2	yes	no	Tests with collection scheme to measure splashed flow from the SG
400	-	2	yes	no	
600	-	2	yes	no	
1200	-	2	yes	no	
TBD	-	2	yes	no	
TBD	-	2	yes	no	

TABLE 5

FREE JET TESTS/DATA REQUIREMENTS

Flow	Number of High-Speed Movies	Minimum Number of Still Photos	CDAS on	Video Tape
200	2	5	yes	no
400	2	5	yes	no
600	2	5	yes	no
800	2	5	yes	no
1200	2	5	yes	no
1200 - 200	-	-	yes	yes

FIGURE 1 - DRAIN TEST FACILITY


SIDE VIEW

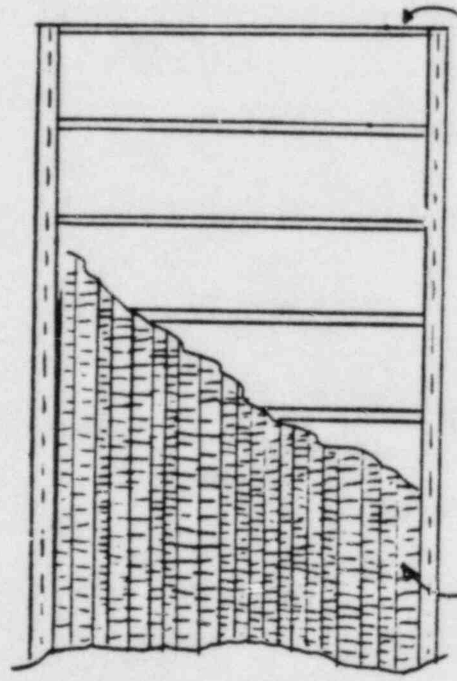
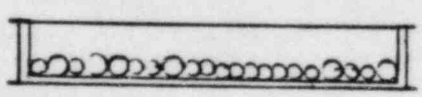

CLOSED END (TYP)
~6"
FULL TRAY
TOP VIEW

FIGURE 2


CABLE TRAY (1", 1¼", & 1½
~2' WIDE
~4" HIGH
END VIEW

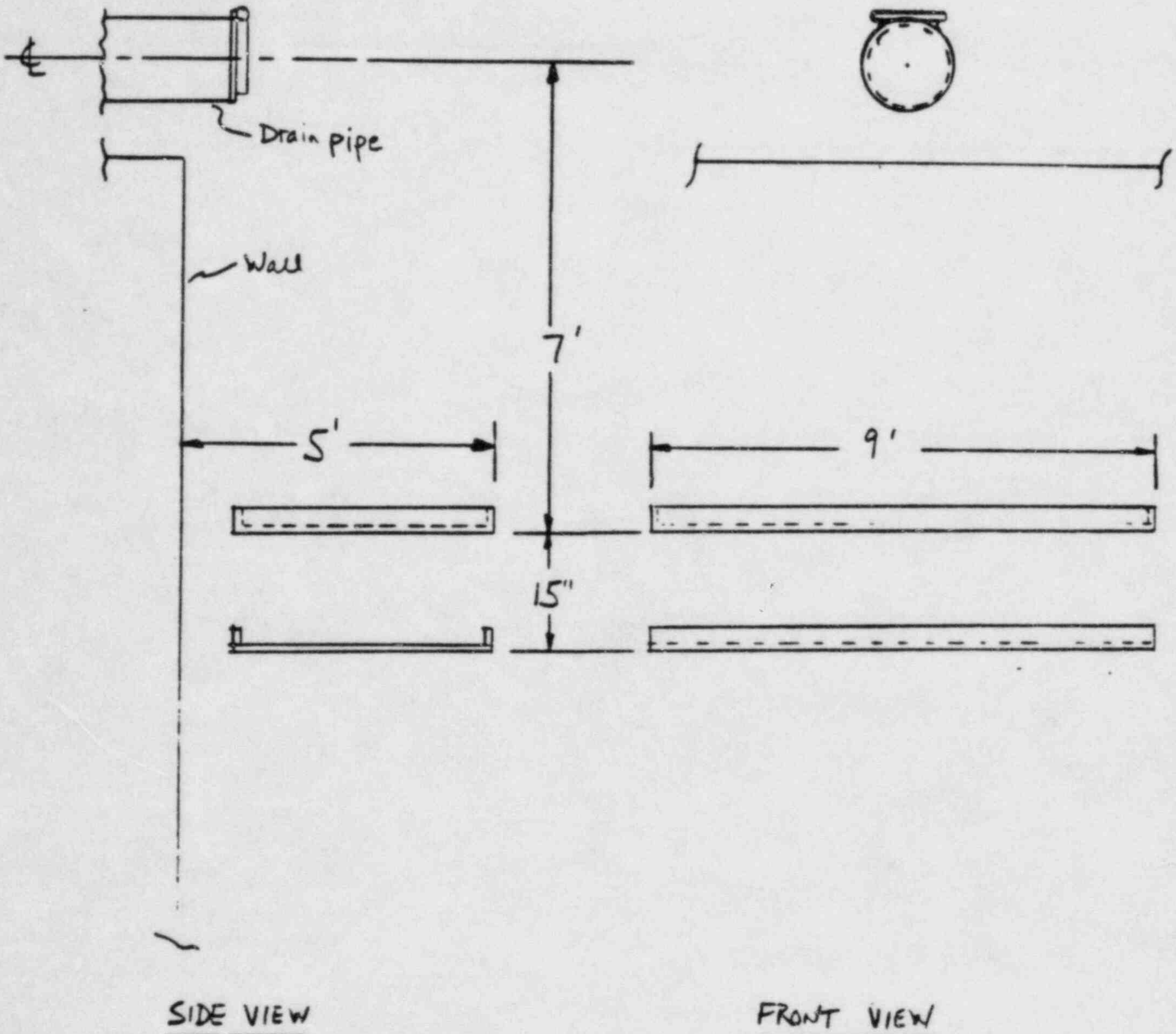


FIGURE 3 - CABLE TRAY ARRANGEMENT

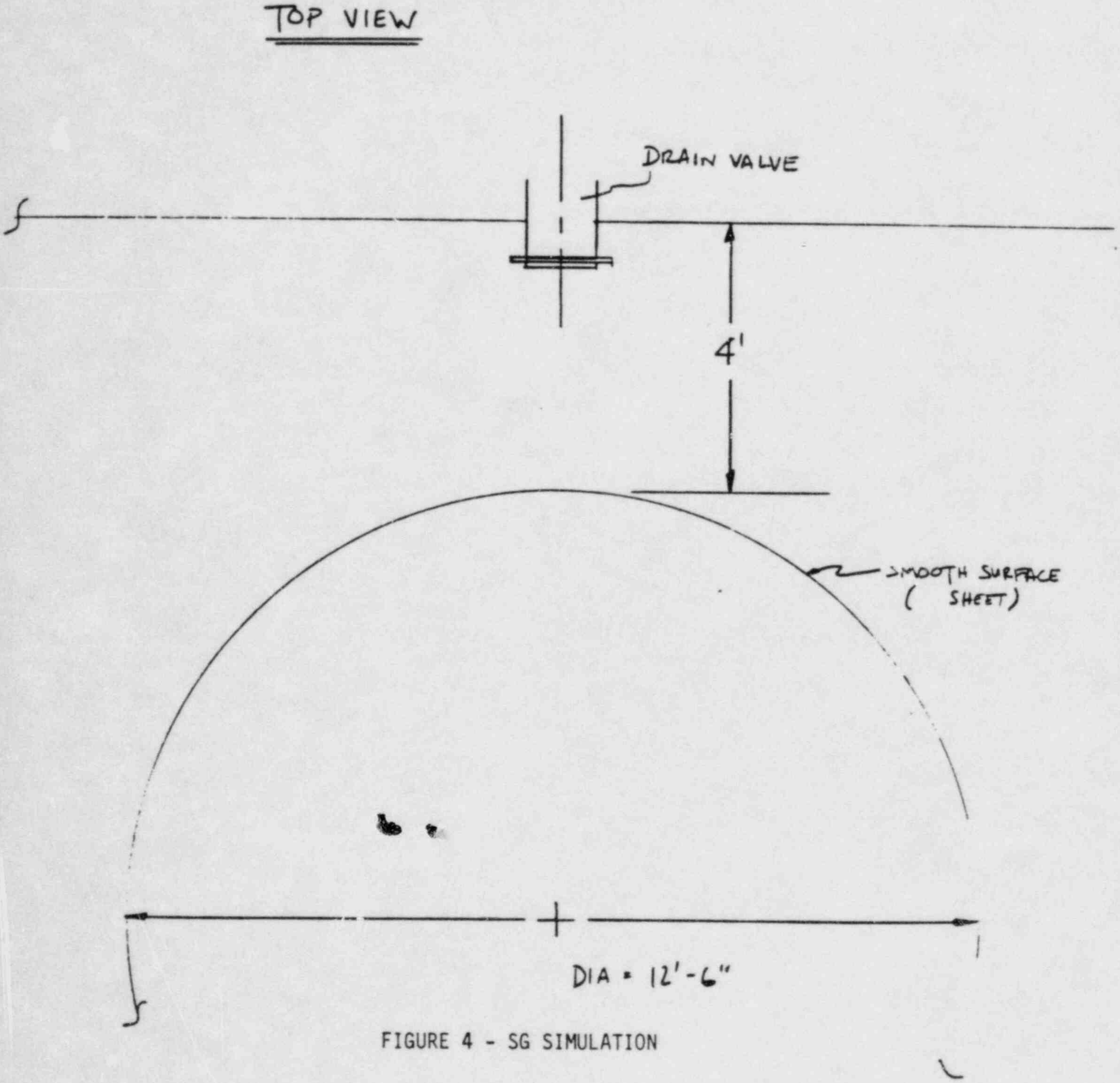


FIGURE 4 - SG SIMULATION

Attachment 4
Catawba Nuclear Station

Preliminary Fracture Mechanics Evaluation

Currently, the environmental analysis for the MSLB in the Catawba containment is performed by postulating breaks in which catastrophic pipe failure is assumed. More realistic estimates of crack opening area and the resulting thermal and mechanical loads can be obtained through application of fracture mechanics techniques. A scoping study has been carried out by Westinghouse for in-containment MSLB's and preliminary results obtained indicate that a non-mechanistic pipe break will not occur in the main steam line.

The purpose of this scoping study was to show that a circumferential flaw larger than any that would be present in the main steam lines will remain stable when subjected to the worst combination of plant loadings. The flaw stability criteria for the analysis examined both the global and local stability. The global analysis was carried out using the plastic instability method, based on traditional plastic limit load concepts but accounting for strain hardening and taking into account the presence of a flaw. The local stability analysis was carried out for a postulated 10 inch long through-wall circumferential flaw. The objective of the local analysis was to show that unstable crack extension will not result for the postulated flaw. The crack opening area resulting from faulted load was calculated for the 10 inch flaw using simplified analysis techniques.

The following results were obtained from the above evaluation:

a. Limit moment calculations indicated that the critical flaw size (beyond which the flaw is unstable) would be <u>greater than</u> the pipe diameter.

b. A postulated 10 inch long through-wall circumferential flaw will remain stable when subjected to maximum faulted load of less than 20 ksi.

c. Available fatigue crack growth results for the main steam line of typical PWR plants indicate no significant crack growth due to the design transients.

d. The crack opening area is estimated to be about 0.2 in^2. If a safety factor of 10 is used, the area would be about 2 square inches.

From these results it is judged that it would be demonstrated by fracture mechanics analysis that catastrophic pipe breaks in the main steam line would not occur. Westinghouse systems evaluation have shown that for crack areas less than 0.1 ft^2 (14.4 in^2), tube bundle uncovery will not occur. Therefore no superheated steam would be generated and original equipment qualification temperature envelopes would not be exceeded.

Attachment 5
Catawba Nuclear Station

IE Lower Containment Equipment Required for Steamline Break Inside Containment

Criteria Used to Identify Required Equipment:

1. Hot standy - Safe Shutdown

Active valves required to maintain the Reactor Coolant System (RCS) pressure boundary must position correctly and remain in that position. Maintain RCS pressure using pressurizer heaters or charging/safety injection (SI) pump cycling. Proceed to cold shutdown after necessary instrumentation is available (operable and verified).

2. Steamline Rupture Assumptions

FSAR single failure assumed.

3. Containment Isolation

Containment isolation valves are required that would result in a direct radioactive release to the environment or loss of refueling water storage tank level. Containment isolation valves that also provide RCS or Steam Generator (SG) pressure boundary isolation are required.

4. Sump recirculation

Switchover to containment sump recirculation is not required for recovery of plant.

IE Equipment Locations:

See Table 1 for a listing of IE equipment located in the lower containment (valves and instrumentation) required for main steamline break inside containment. Equipment elevation, radius, and azimuth are provided. Equipment locations are also marked on attached reactor building plans with elevations noted under valve or instrument number.

IE Lower Containment Equipment Required:

1. Valves required to prevent direct release of containment atmosphere

VQ2A VQ16A	Containment Air Release and Addition Isolation
VP7A VP9A VP19A VP15A VP17A	Containment Purge Isolation

The Containment Purge and the Containment Air Release and Addition Valves are normally closed. The inside containment isolation valves are located in dead ended compartments. The redundant isolation valves are located outside containment. Valves in both systems receive a containment isolation signal to close. Radiological consequences of loss of all function would be enveloped by the analysis for the steamline break outside containment.

2. Valves required to maintain RCS pressure boundary.

NV123B NV122B	Excess Letdown Isolation
NV11A NV10A NV13A	Letdown Isolation
ND2A ND37A ND1B ND36B	Residual Heat Removal (RHR) Isolation
NM22A NM25A NM3A NM6A	RCS Sample Isolation
NC32B NC34A NC36B	Pressurizer Power Operated Relief (PORV) Valves
NC31B NC33A NC35B	PORV Block Valves
NC250A NC251B NC252B NC253A	Head Vent Isolation

Excess letdown isolation valves are normally closed air operated valves. These valves are used only during startup. The fail closed air operated control valve downstream of the excess letdown heat exchanger serves as a backup to the letdown isolation valves. The control valve is also normally closed.

Letdown isolation valves are normally open air operated valves located in a dead ended compartment and receive a containment isolation signal to close. Air operated valves upstream that close automatically on a pressurizer low level signal serve as a backup to the letdown isolation valves.

RCS sample isolation valves, located in dead ended compartments, receive a containment isolation signal to close. One of the two isolation valves is normally closed. The two lines are also isolated by a redundant valve located outside containment. Pressurizer sample isolation valves are normally closed and receive a containment isolation signal to close. The two lines are also isolated by a redundant valve located outside containment. Loss of the inside and outside isolation valves is acceptable unless the non-safety grade tubing to the sample lab is also lost.

Head vent isolation valves are normally closed solenoid valves. Valves NC252B and 253A have power removed during normal operation. Since piping downstream of the head vent valves contains a 3/8" restriction, loss of all head vent valves would result in a RCS leak within the high head charging pump capacity.

The pressurizer power operated relief valves and block valves are located near the top of the pressurizer enclosure (dead ended compartment). The PORVs are normally closed, fail closed air operated valves. The steamline break will not cause these valves to open. The motor operated block valves are used to isolate the PORVs if required.

The residual heat removal isolation valves are normally closed motor operated valves. The RHR suction piping consists of two parallel lines with two isolation valves in series in each line. One of the two valves in each line is in a dead ended compartment. Also, one of the two valves in each line has power removed during normal operation.

3. Valves required to prevent reactor coolant release outside containment (not RCS pressure boundary concern)

NC54A	Pressurizer Relief Tank Isolation
WL450A	Reactor Coolant Drain Tank to Waste Gas Iaolation
WL805A	Reactor Coolant Drain Tank to Recycle Holdup Tank Isolation
NV89A	Excess Letdown and Seal Return Isolation

Valves used to isolate the pressurizer relief tank, the reactor coolant drain tank and the excess letdown and seal return line are located in dead ended compartments and receive a containment isolation signal to close. Redundant isolation valves are located outside containment in each flow path. Loss of function of both valves would not result in loss of fluid without the additional failure of non-safety piping and valves in the auxiliary building.

4. Valves required to maintain Steam Generator (SG) pressure boundary

BB56A
BB19A
BB60A
BB8A

Steam Generator Blowdown Isolation

NM187A
NM197B
NM207A
NM217B
NM190A
NM200B
NM210A
NM220B

Steam Generator Sample Isolation

Isolation of SG blowdown and SG sampling is required to prevent loss of feedwater from the steam generators. Each line contains a normally open motor operated valve inside containment that receives a containment isolation signal to close. Redundant isolation valves are located outside containment. SG blowdown is also isolated in the turbine building by closing the control valves in the blowdown lines. Sample valves and blowdown valves close on an automatic start signal for the Auxiliary Feedwater Pumps (AFP). Failure of the inside and outside containment isolation valves would not cause a loss of inventory without the additional loss of non-safety piping and valves outside containment.

5. IE instrumentation inside containment required for actuation

NCPT5150
NCPT5160 } Pressurizer Pressure Transmitter
NCPT5170

All sensors used to provide actuation signals for the steamline break are located outside containment except pressurizer pressure transmitters. These transmitters are located in dead ended compartments. The signal from this transmitter is used to generate (P11) interlock and is only applicable for small breaks at low power levels.

6. IE instrumentation inside containment required for Post Accident Monitoring System (PAMS)

NCLT5150
NCLT5160 } Pressurizer Level Transmitter
NCLT5170

NCRD5850	Loop 1 HL Wide Range Temperature
NCRD5860	Loop 1 CL Wide Range Temperature
NCRD5870	Loop 2 HL Wide Range Temperature
NCRD5880	Loop 2 CL Wide Range Temperature

Core Exit Temperature

Containment Radiation

Pressurizer level transmitters are located in dead-ended compartments.

Incore Thermocouple System (ITS) cables and connectors are the only parts of the ITS exposed to the accident environment inside containment. The cables and connectors have been tested and qualified to 460°F which is documented under Combustion Engineering Report No. 17682-CCE-SR80-1, Rev. 00.

The only components of the RVLIS system exposed to the inside containment accident environment are the MINCO RTDs and cables. The RTDs and cables have been tested and qualified to 420°F which is documented in WCAP-8687, Supp. 2, E-42A.

The Hi-range Radiation Monitors have been tested and qualified for a LOCA. Although the monitors are not qualified for the MSLB high temperatures, the capability is present to obtain equivalent information on long term containment radiation levels through qualified sample systems or other radiation readings correlated to the containment. It should be noted that the high range radiation monitoring instrumentation does not serve as a basis for long-term operator actions.

The wide range RCS temperature utilizes RdF Corp. RTDs and associated cables. They have been tested and qualified to 420°F. It should be noted that the normal continuous operating temperature of the RTDs is approximately 600°F which exceeds the postulated accident temperature peaks for all DBEs.

Table 1 (Page 1)

LOWER CONTAINMENT CLASS 1E

VALVES REQUIRED FOR MAIN STEAMLINE BREAK INSIDE CONTAINMENT

SYSTEM	TYPE OPERATOR	DUKE VALVE ID	MFG/MODEL	LOCATION	ELEV.	RAD	AZ	ACTUAT. SIGNAL	FUNCTION	COMMENTS
Containment Air Release And Addition	MOV	VQ2A		IC/DE	636	53	266	T	Cont. Pr. Bdr. Isol	Valves normally closed. Valves receive cont. isol signal to close. Redundant isolation valves located outside containment.
	MOV	VQ16A		IC/DE	559	50	165	T	Cont. Pr. Bdr. Isol	
Containment Purge	AOV	VP7		IC/DE	587	58	242	T	Cont. Pr. Bdr. Isol	Valves normally closed. Valves receive cont. isol signal to close. Redundant isol. valves located outside containment.
	AOV	VP9A		IC/DE	587	58	307	T	Cont. Pr. Bdr. Isol	
	AOV	VP19A		IC/DE	590	58	75	T	Cont. Pr. Bdr. Isol	
	AOV	VP15A		IC/DE	590	58	59	T	Cont. Pr. Bdr. Isol	
	AOV	VP17A		IC/DE	890	58	101	T	Cont. Pr. Bdr. Isol	

IC - Inside Containment

DE - Dead Ended Compartment

T signal - Containment Isolation Signal

MOV - Motor Operated Valve

AOV - Air Operated Valve

Table 1 (Page 2)

LOWER CONTAINMENT CLASS 1E

VALVES REQUIRED FOR MAIN STEAMLINE BREAK INSIDE CONTAINMENT

SYSTEM	TYPE OPERATOR	DUKE VALVE ID	MFG/MODEL	LOCATION	ELEV.	RAD	AZ	ACTUAT. SIGNAL	FUNCTION	COMMENTS
RCS	AOV	NC32B	Valcor	IC/DE	635	38	107		Prz. relief valve	Valves normally closed.
		NC36B	Valcor	IC/DE	635	37	101		Prz. relief valve	Valves located in pressu-
		NC34A	Valcor	IC/DE	635	37	105		Prz. relief valve	rizer compartment.
	MOV	NC31B	Limitorque	IC/DE	635	35	107		Prz. block valves	Block valves used to
		NC33A	Limitorque	IC/DE	635	34	105		Prz. block valves	isolate Prz power operated
		NC35B	Limitorque	IC/DE	636	35	101		Prz. block valves	relief if required.
	MOV	NC54A	Limitorque	IC/DE	559	50	111	T	PRT iso	Normally closed, receives cont. isolation signal to close. Redundant isolation valve located outside containment.
CVCS	MOV	NV89A	ROTORK	IC/DE	559	56	165	T	Excess letdown & seal return isolation	Receives cont. isolation signal to close. Redundant isolation valve located outside containment
	AOV	NV123B	Valcor NAMCO	IC/MC	553	20	230		Excess letdown iso	Valves normally closed.
	AOV	NV122B	Valcor	IC/MC	558	21	230		Excess letdown iso	
	AOV	NV11A	Valcor	IC/DE	579	49	116	T	Letdown iso	Letdown isolation valves
		NV10A	Valcor	IC/DE	581	48	115	T	Letdown iso	receive cont. isol. signal
		NV13A	Valcor	IC/DE	579	47	116	T	Letdown iso	to close.
RHR	MOV	ND2A	ROTORK	IC/DE	557	50	176		HL to RHR	RHR isolation
	MOV	ND37A	ROTORK	IC/DE	567	50	184		HL to RHR	valves normally closed.
	MOV	ND1B	ROTORK	IC/MC	569	25	176		HL to RHR	Power removed from 1of 2
	MOV	ND36B	ROTORK	IC/MC	564	24	181		HL to RHR	valves in each flow path.
WPS	MOV	WL450A	Limitorqure	IC/DE	582	56	237	T	RCDT to Waste Gas System	Valves receive cont. isol. signal to close. Redun-
	MOV	WL805A	ROTORK	IC/DE	569	55	242	T	RCDT to RHT	dant isolation valve located outside containment.

Table 1 (Page 3)

LOWER CONTAINMENT CLASS 1E

VALVES REQUIRED FOR MAIN STEAMLINE BREAK INSIDE CONTAINMENT

SYSTEM	TYPE OPERATOR	DUKE VALVE ID	MFG/MODEL	LOCATION	ELEV.	RAD	AZ	ACTUAT. SIGNAL	FUNCTION	COMMENTS
Head Vent	MOV	NC250A	Limitorque	IC/MC	600	25	38		RV Head Vent	Valves normally closed. Valves 1NC252B and 1NC253A have power removed during normal operation. Common line downstream of valves contains 3/8" restriction.
	MOV	NC251B	Limitorque	IC/MC	600	25	40		RV Head Vent	
	MOV	NC252B	Limitorque	IC/MC	600	26	38		RV Head Vent	
	MOV	NC253A	Limitorque	IC/MC	600	27	40		RV Head Vent	
RC Sample	MOV	NM22A	Limitorque	IC/MC	576	41	188	T	RC Sample	One of two valves normally closed. Both receive cont. isol. signal to close. Redundant isolation valve located outside cont.
	MOV	NM25A	Limitorque	IC/MC	576	31	193	T	RC Sample	
	MOV	NM3A	Limitorque	IC/DE	583	46	140	T	RC Sample	Normally closed except during Prz sampling. Both receive cont. isol. signal to close. Redundant isol. valve located outside cont.
	MOV	NM6A	Limitorque	IC/DE	581	45	129	T	RC Sample	
SG Blowdown	MOV	BB56A		IC/MC	575	36		T	SG Blowdown	Valves receive cont. isol. signal to close. Redundant isolation valves located outside cont. Valves close on auto start of Aux Fd Pmp control valves in Turbine Building also close on auto start of Aux. Fd Pump to isolate blowdown.
	MOV	BB19A		IC/MC	577	38		T	SG Blowdown	
	MOV	BB60A		IC/MC	577	38		T	SG Blowdown	
	MOV	BB8A		IC/MC	575	38		T	SG Blowdown	
SG Sample	MOV	NM187A	ROTORK NA-1	IC/MC	562	41	45	T	SG Sample	Valves receive cont. isolation signal to close. Redundant isolation valve located outside cont.
	MOV	NM197B	ROTORK NA-1	IC/MC	565	38	163	T	SG Sample	
	MOV	NM207A	ROTORK NA-1	IC/MC	565	38	218	T	SG Sample	
	MOV	NM217B	ROTORK NA-1	IC/MC	562	40	334	T	SG Sample	
	MOV	NM190A	ROTORK NA-1	IC/MC	562	37	21	T	SG Sample	
	MOV	NM200B	ROTORK NA-1	IC/MC	556	38	180	T	SG Sample	
	MOV	NM210A	ROTORK NA-1	IC/DE	559	44	212	T	SG Sample	
	MOV	NM220B	ROTORK NA-1	IC/DE	559	49	229	T	SG Sample	

Table 1 (Page 4)

LOWER CONTAINMENT CLASS 1E

INSTRUMENTATION REQUIRED FOR MAIN STEAM BREAK INSIDE CONTAINMENT

SYSTEM	DUKE ID	LOCATION	ELEV	RAD	AZ	SENSOR	FUNCTION
RCS	NC PT 5150	IC/DE	569	58	91	Prz Press	P-11 Interlock
	NC PT 5160	IC/DE	569	58	98	Prz press	P-11 Interlock
	NC PT 5170	IC/DE	569	58	102	Prz Press	P-11 Interlock
	NC LT 5150	IC/DE	570	58	91	Prz Level	PAMS
	NC LT 5160	IC/DE	570	58	98	Prz Level	PAMS
	NC LT 5170	IC/DE	571	102	102	Prz Level	PAMS
	NC RD 5850	IC/MC	567	20	20	Loop 1HL Wide Rang Temp	PAMS
	NC RD 5860	IC/MC	567	28	51	Loop 1CL Wide Rang Temp	PAMS
	NC RD 5870	IC/MC	567	18	160	Loop 2HL Wide Rang Temp	PAMS
	NC RD 5880	IC/MC	567	28	124	Loop 2CL Wide Rang Temp	PAMS
	Incore Thermocouple	IC/MC	Reactor Core			Core Exit Temp	PAMS
	RVLIS Thermocouple for Temp Compensation	IC/MC	See attached RVLIS drawing			RVLIS Temp	PAMS