50-(237) 249

Regulatory File Cy.

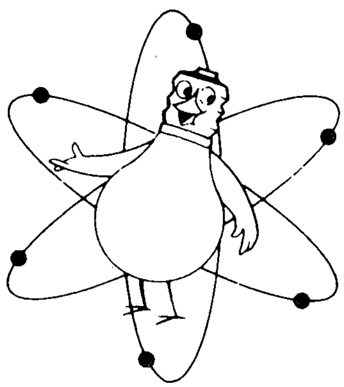
Received w/Ltr Dated 8-18-69

SUPPLEMENTARY INFORMATION

DRESDEN NUCLEAR POWER STATION UNITS 2 AND 3

AMENDMENT NUMBER 17 FOR UNIT 2 AND AMENDMENT NUMBER 18 FOR UNIT 3

AMENDS. 17 & 18



Commonwealth Edison Company

2645-A

50-237-249

Regulatory File Cy.

Received w/Ltr Dated 8-18-69

TABLE OF CONTENTS

1-Supplemental Information - ROD BLOCK MONITOR

"Provide a detailed comparison of the Rod Block Monitor System to IEEE 279 indicating all aspects wherein the Rod Block Monitor System does not comply with the code.

"Provide an evaluation of the improvement in the overall plant safety factor which would be obtained if the Rod Block Monitor System were revised to meet IEEE 279."

SUMMARY

A detailed analysis has been made of the functional requirements of the Rod Block Monitor System. The results of this assessment places in proper perspective the requirements of the Rod Block Monitor System as compared to the requirements of reactor protection systems. The analysis clearly reveals that (1) the Rod Block Monitor System performs no primary reactor protection function, and (2) the system as designed with redundancy of channels and equipment for testing and maintenance will reliably and accurately serve its intended functions, i.e., to avoid local fuel damage as a result of extreme abnormal transients. The contribution of the Rod Block Monitor System to the protection of health and safety of the public, is insignificant. It is thus concluded that designing the Rod Block Monitor to the requirements of IEEE 279 would provide no substantial addition to the protection of the public health and safety.

COMPARISON TO IEEE 279

A comparison has been made of the Rod Block Monitor system conformance to IEEE 279. The following list represents various aspects wherein the Rod Block Monitor system does not comply with Reactor Protection system grade separation, redundancy, and isolation.

1. Two redundant RBM channels are provided. These channels are located in the same cabinets and thus are not separated or isolated.

2. The rod selection information including the rod selection pushbutton is not redundant.

3. A single set of taps are provided on the flow measurement venturi in each recirculation loop.

4. The flow sensing lines used for each rod block monitor are not separated or isolated.

5. The flow signal transducers for the two rod block monitors are not separated or isolated.

6. The LPRM amplifier signal inputs and outputs are neither separated nor isolated from each other.

7. The APRM reference signals which are inputted to each RBM channel through the LPRM Select Matrix are independent, but are located in the same cabinets and thus not separated.

8. The independent RBM level readouts and status displays are neither separated nor isolated.

9. The independent rod block signals to the Reactor Manual Control System circuitry are neither separated nor isolated.

The RBM system does contain some redundancy is that two independent RBM channels are provided. Each of these channels independently blocks rod withdrawal, receives inputs from different LPRM's and has its own power supply.

A detailed analysis was performed on the RBM system to determine under what conditions the RBM would be instrumental in preventing reduction of thermal margins. This detailed analysis, showing the most limiting case, was presented in the Hatch Nuclear Plant, Amendment 6, Docket 50-321. It was shown that if both RBM's fail concurrently with the operators failure to obey procedures by withdrawing the highest worth rod from the most adverse control rod pattern which can be postulated to exist, and by ignoring the various series of warning alarms, there would be no significant radiological consequences. In fact, the most limiting release would be within the limitations specified by 10CFR20.

If reactor power is less than 75% with any permitted rod pattern, no rod withdrawal error coincident with RBM system failure can result in a MCHFR of less than 1.0.

Consider then the normal operation including constant changes in power level to account for day to day load requirements. In the normal rod patterns which are used, the withdrawing of a maximum worth control rod concurrently with the failure of the RBM system would not result in a MCHFR less than 1.13.

Preliminary investigations have been made toward redesigning the Rod Block Monitor system to meet the intent of IEEE 279. The investigations show that a redesign which permits the RBM's to remain in the same cabinet, but which separates each RBM with a horizontal barrier, is the most feasible solution. This redesign would include:

1. Change the physical location of the RBM's in Panel 937 and add fire resistant barrier.
2. Rewire "selected" signal leads in Panel 937.
3. Rebuild LPRM inputs to RBM selection matrices. Add isolation amplifiers in the LPRM's.
4. Relocate the recirculation flow transmitters. Add flow reference signal isolation features in flow reference equipment. Add redundant taps on the recirculation flow nozzles.
5. Separate the APRM reference signals to the RBM's.
6. Separate the RBM outputs.
7. Rewire the RBM signals to the Reactor Manual Control System from Panel 928.

On each of the above changes electrical separation must be maintained where rewiring is involved.

EVALUATION

The Rod Block Monitor System is designed to give an additional depth of automatic protection against the single rod withdrawal error so that MCHFR in the surrounding fuel bundles is not significantly reduced below 1.9. Of first interest is the depth of protection against the event before the RBM is called upon to act. Table 1 summarizes the various errors and malfunctions pertinent to RBM operation.

(1) An abnormal and special pattern of rods must be set up such that at full power the central rod of the pattern is fully inserted and the surrounding rods are fully or substantially withdrawn. To set up this control rod pattern requires several operator errors, violation of normal operating procedures, and ignoring of several alarms. (For purposes of suppressing the flux around a certain fuel bundle, rod patterns of this type might be established under the direction and authorization of the plant Nuclear Engineer. However, it is expected that such a rod pattern would not be established on more than a few days during the 40 year plant life).

(2) The plant operator must erroneously select <u>and</u> withdraw the single high worth rod.

(3) He must erroneously continue to withdraw the single high worth rod in spite of all of the following warnings:

a) A flashing light
b) A sounding horn
c) An individual LPRM trip light display on each of two panels.
d) A Four-Rod Display with numerous meters showing high flux levels on the affected channels
e) Repeated audible alarms on the computer alarm typewriter

All of the above events would have to occur before the Rod Block Monitor is called into service. The RBM utilizes the same information that is available to the operator and automatically inhibits rod withdrawal before preselected flux levels are exceeded. Many of the features of the RBM are duplicated, primarily so that one channel at a time can be by-passed to allow adequate time for maintenance and repair without encroaching on either the availability of the rod block action availability. The redundancy in the RBM is not as complete as that in systems of primary safety significance; some of the channels and signals do not have the same degree of separation and isolation as is found in the reactor protection system.

Qualitatively, it is expected that the need for an RBM signal will be an extremely rare event. True, a human operator could, theoretically set up a situation wherein the RBM could be called upon the block withdrawal of an erroneously, selected control rod, but this event is not of the same probability as such expected transients as a turbine trip, a loss of off-site power, or a loss of feedwater flow.

Failure of the RBM is also expected to be a rare event. It is largely redundant and thus requires multiple failures or at least highly selective single failures to cause system failure.

CONCLUSION

The significant facts pertinent to the functioning of the RBM System are as follows:

1. For any rod withdrawal error under normal conditions, complete failure of the RBM does not result in MCHFR of less than 1.0.
2. For any rod withdrawal error from any permitted rod pattern (normal or abnormal) at power levels below 75%, complete failure of the RBM system does not result in MCHFR of less than 1.0.
3. For the worst possible rod withdrawal error, a case involving multiple operator errors and multiple equipment failures, complete failure of the RBM System results at worst by consequences which are within the limitations of 10CFR20.
4. The RBM System has two channels which have limited redundancy and independence.

Thus, it is concluded that no substantial additional improvement to the protection of the public health and safety is gained by designing this system, which is not a primary reactor protection system, to the requirements of IEEE 279.

D-2, 3

TABLE 1

ASSUMED OPERATING ERROR AND MALFUNCTION CASES

	Normal Rod Pattern Cases			Abnormal Rod Pattern* Cases			
	A	B	C	A	B	C	D
OPERATOR ERRORS							
1. Any Single Operator Error	1						
2. A specific or combination of Operator Errors							
a. Begin withdrawal of wrong rod			1	1	1	1	1
b. Ignore - LPRM anunciator			1	1		1	1
- 3 LPRM indicators			1	1		1	1
c. Ignore off-gas alarm	NA	NA	NA	NA	NA	1	
TOTAL OPERATOR ERRORS	1	0	3	3	1	4	3
EQUIPMENT MALFUNCTIONS							
1. Any single equipment malfunction		1					
2. A specific or combination of equipment malfunctions							
RBM Failure(s)			(1) 1-3		(1) 1-3	(1) 1-3	(1) 1-3
Air Ejector Monitor Failure	NA	NA	NA	NA	NA		
TOTAL EQUIPMENT MALFUNCTIONS	0	1	1-3	0	1-3	1-3	1-3
TOTAL ERRORS AND MALFUNCTIONS	1	1	4-6	3	2-4	5-7	5-7
RESULTS							
Number of fuel rods affected	0	0	0	0	0	(2) 28/168	(2) 28/168
Core Thermal Margin (MCHFR)	>1	>1	>1	>1	>1	<1	<1
10CFR20 Limits exceeded	No	No	No	No	No	No	No

Footnotes:

(1) Various RBM failures have been postulated; in many of these cases 3 or more failures are required to prevent a rod block.

(2) Possible condition if RBM failure results in block being upscaled, with operation at 83% power (28). Maximum core effect at 100% power (168).

* Rod patterns which could produce the results postulated in these paths are highly unusual and would be established only on a few occasions during the 40 year expected plant operating life.

2-Supplemental Information - APRM FLOW REFERENCE SCRAM

> "Provide a detailed comparison of the Flow Reference Scram Components and Circuitry indicating all aspects wherein the Flow Reference Scram Equipment does not comply with the code."
>
> "Provide an evaluation of the improvement in the overall plant safety factor which would be obtained if the Flow Reference Scram Equipment were revised to meet IEEE 279."

A comparison has been made of the Flow Reference Scram Components and circuitry placement and separation to assess their conformance to IEEE 279. The following list represents various aspects wherein the Flow Reference Scram Components and Circuitry do not comply with Reactor Protection System grade separation, redundancy, and isolation based on our interpretation of IEEE 279:

1. A single set of taps are provided on the flow measurement venturies.
2. The flow sensing lines are not separated and isolated.
3. The flow signal pressure transducers are not separated and isolated.
4. Only one flow signal is provided as a reference to each Trip System:
   a. There is only one set (2) of flow transmitters per Trip System
   b. There is only one set (2) of Square Rooters per Trip System
   c. There is only one summer per Trip System
   d. There is only one Flow Converter, Flow Comparator and Power Supply per Trip System
5. Flow signal inputs are not separated and isolated.
6. Flow Converter outputs (reference signals) are not separated and isolated.
7. Flow Converter outputs (upscale and Inoperative Trips) are not redundant.

The Flow Reference Scram Equipment does contain some redundancy in that the final output to the APRM's is compared by the flow comparators (one in each Trip System) and an alarm and rod block is provided when the difference between the two reference signals exceeds about 10% of flow.

The various transients which the reactor system may encounter over its lifetime are listed throughout the FSAR. The safety protection system components which terminate these transients were designed to prevent the reactor core from reduction in thermal margins to less than MCHFR of 1.0. As such, the safety protection equipment was designed in accordance with IEEE 279.

The reactor protection system fixed flux scram (120%) has been shown to adequately protect against reduced thermal margins. There is no transient which requires the Flow Reference Scram to prevent reduction of the thermal margins. Thus, the overall plant safety does not improve with the addition of the Flow Reference Scram and the health and safety of the public is not enhanced.

3-Supplemental Information - CONTROL ROOM VENTILATION SYSTEM

"Determine the capability of the control room ventilation system to minimize the inhalation dose such that it is acceptable when considering the TID 14844 source term".

Fission products from LOCA in Units 2-3 will be released from the top of the 310' chimney. Should these fission products reach ground level during a brief atmospheric fumigation condition, the control room ventilation system for the three control rooms can be isolated from outside air intake. The ventilation systems are designed to operate on a closed cycle.

Should an accident occur during an atmospheric conditions which would direct radioactivity toward the control room ventilation inlet duct, an extremely small amount of radioactive gases will enter the control room area. There are radiation monitors located at the east end of Unit 1 Control Room and at the west end of Unit 2/3 Control Room both having ranges of 0.01 mr/hr to 100 mr/hr which would actuate an alarm indicating high radiation in the control room. The operator would manually switch the ventilation system to its recirculation mode, thus preventing further intake of radioactive gases.

The control room ventilation system in its normal mode of operation draws 10% of its air from outside the control room building, the remaining 90% from inside the control room building. The total air is drawn through a Pliotron filter which removes dust particles, cooling coils or heating coils cool or heat the air as required, and blowers which circulate the air to the control room. There is no design feature which removes or otherwise reduces the activity level of the radioactive gases which could enter the control room via the control room ventilation system.

An analysis has been performed using worst case meteorological conditions(i.e., Pasquil type A),the TID 14844 source term, a containment leak rate of 1% per day (AEC assumption) and a filter efficiency of 90% in the standby gas treatment system (AEC assumption). A low wind velocity with direction toward the inlet duct of the control room ventilation system is also assumed. Intake through the control room ventilation for a period of 5 minutes was assumed as representative of the time needed by the operator subsequent to high level alarm from the control room area monitors to terminate use of outside air in the ventilation flow. The calculated dose is much less than the criterion of 0.5 rem/8 hours. This is mainly a result of the favorable high elevation of the release from the stack and the small fraction of the cloud entering the ventilation even under worse case conditions.

Supplemental Information - STACK MONITORING CAPABILITY

> "Determine the relative capability of the stack gas monitoring and the standby gas treatment system monitoring to read discharges during a TID 14844 release."

The stack monitors are intended to audit the quantity of radioactive material released via the stack during normal operation. This audit is a gross measurement since the effluent is predominantly noble gases and is to verify compliance with 10CFR20. To accomplish this, audit is made of the stack flow consisting of radiogases in the air ejector offgas (approximately 230 cfm, see page 9.2-2 of FSAR) mixed with ventilation air (approximately 200,000 cfm, see page A-3-9 of the FSAR). Sampling is done in accordance with good engineering design practice, i.e., the sample line is located at a height on the stack which is 10 times the stack diameter. The instrument measures concentration which is calibrated in terms of μCi/sec of effluent discharge.

In the event that operation of the standby gas treatment system is required, normal offgas flow would be terminated and normal ventilation may be terminated depending on availability of off-site power. The flow from the gas treatment system is 4000 cfm which is directed to the plant stack near its base. Thus, the stack gas monitor sampling line could sample the gas treatment flow; however, since the monitor is designed for sampling a much larger flow, with quite low concentrations, the sampling of this small concentrated flow (if normal ventilation is stopped) could be beyond the instrument sensitivity. As can be seen from Chapter 14 of the FSAR, the radiogas release rates during accident conditions could potentially be .056 Ci/sec (see page 14.215 of FSAR) diluted in 4000 cfm or a concentration of approximatley 8×10^{-4} Ci/ft^3 if mixed uniformally. This may be compared to instrument sensitivity of 20 Ci/sec maximum normal release rate as specified in the tech specs diluted into 200,000 cfm. (See analysis below) or a concentration of 6×10^{-3} Ci/ft^3. Thus, the concentration during the accident as described in the FSAR is less than the maximum during normal operation. If an adequate sample of this low flow is made, the monitor will read on scale during the accident as described in the FSAR.

Instrument limitation is evident if a more conservative accident model is assumed as is done by the AEC staff (using TID fission product source). In such an event, even larger fission product release is presumed and would certainly be beyond the instrument sensitivity. An additional reason why the stack monitor is unsuited to measurement of the accident effluent is that the fission product mixture for such an effluent is energetically quite varied relative to the normal noble gas mixture in the offgas.

Thus, it can be concluded that the stack monitoring system which is intended for normal audit will not be suitable to assure proper monitoring of the gas treatment effluent during accident conditions if the TID assumptions are used. The monitor will efficiently monitor up to about 1% of the TID source term. However, as noted below, the reactor building ventilation monitor which initiates the standby gas treatment system will remain on scale during a release as large as the TID source.

Since the effluent is almost entirely noble gases the gross measurement at the standby gas treatment system is representative of a gross measurement of that going out the stack. The filter downstream of the standby gas treatment monitor affects only the iodine and particulate material which are a small part of the total radioactive material content and not detected individually by a gross measurement.

Analysis:

The following experimentally determined factors for use in determining equipment sensitivity were performed at Vallecitos.

Stack Flow Rate: 2×10^5 Ft^3/min

Sample Flow Rate: 2 Ft^3/min

Activity : 10 ci/sec

The readout of the monitoring instrument which is the same as the Dresden instrument was 10^5 c/s.

I. Using the above factors the activity per unit volume in stack can be determined as follows:

$$A_v = \frac{10 \text{ ci/sec} \times 60 \text{ sec/min}}{2 \times 10^5 \text{ Ft}^3\text{/min}} = 3 \times 10^{-3} \text{ ci/Ft}^3$$

II. From the information in the above Equipment sensitivity/Activity/Volume is

$$= \frac{10^5 c/s}{3 \times 10^{-3} \frac{Ci}{Ft^3}} \qquad = \frac{3.3 \times 10^7 c/s}{ci/Ft^3}$$

III. Using the above calculations the ability of the stack monitor to measure the effluent during accident conditions has been determined. Monitor capability is 2×10^5 counts/sec equivalent to 20 Ci/sec in a flow of 200,000 cfm.

Reading for Standby Gas w/TID release

$$A_v = \frac{25 \text{ Ci/sec} \times 60}{4.0 \times 10^3 \text{ Ft}^3/\text{min}}$$

$$A_v = 3.9 \times 10^{-1} \text{ ci/Ft}^3$$

$$\text{Reading} = 0.39 \text{ ci/Ft}^3 \times 3.3 \times 10^7 \frac{c/s}{ci/Ft^3} = \underline{1.3 \times 10^7 c/s} \text{ --OFF SCLAE}$$

IV. Reading for Loss of Coolant in FSAR

$$A_v - \frac{2.2 \times 10^{-2} \text{ ci/sec} \times 60 \text{ sec/min}}{4.0 \times 10^3 \text{ Ft}^3/\text{min}} = 3.3 \times 10^{-4} \text{ ci/Ft}^3$$

$$\text{Reading} = 3.3 \times 10^{-4} \times 3.3 \times 10^7 \frac{c/s}{Ci/Ft^3} = \underline{1.1 \times 10^4 c/s}\text{--ON SCALE}$$

V. Reading for Refueling Accident in FSAR

$$A_v = \frac{5.6 \times 10^{-2} \times 60}{4.0 \times 10^3} = 8.4 \times 10^{-4} \frac{ci}{Ft^3}$$

$$\text{Reading} = 8.4 \times 10^{-4} \frac{ci}{Ft^3} \times 3.3 \times 10^7 \frac{c/s}{ci/Ft^3} = \underline{2.8 \times 10^4 \text{ c/s}}\text{--ON SCALE}$$

<u>Reactor Building Ventilation Monitor</u>

Trip level

$$9.9 \times 10^{-5} \text{ Ci/Ft}^3 = 11 \text{ mr/hr} = 0.33 \text{ Ci/sec}$$

$$\text{or } \frac{33 \text{ mr/hr}}{\text{Ci/sec}}$$

$$\text{TID} \quad 25 \text{ Ci/sec} \times \frac{33 \text{ mr/hr}}{\text{Ci/sec}} = \underline{830 \text{ mr/hr}}$$

Monitor range 0.1 - 1000 mr/hr Therefore, monitor stays on scale.

Supplementary Information - DRESDEN 1 MAIN STEAMLINE CONTAINMENT ISOLATION VALVE SPECIFICATION AND LEAK RATE TEST INFORMATION

> "Provide a copy of the Dresden 1 main steamline containment isolation valve design specification and the test results of the isolation valve leak rate test for the lifetime of the plant."

Commonwealth Edison has reviewed the records of the main steam line isolation valve leakage tests for Dresden Unit 1. On Dresden Unit 1, there are two main steam lines each having a gate valve serving as an isolation valve inboard of the containment and a turbine stop valve serving as the outboard isolation valve. These tests were done on a yearly basis to satisfy licensing requirements for containment leak rate testing. Through 1965 the leak testing of the main steam line isolation valves was done using water injected between the inboard and outboard valves. The amount of water leakage was measured; and based on this measured water leakage, an equivalent air leakage was calculated. The results of these tests showed no trend in increased leakage as a function of time. It was considered that a water test of these valves was not a good indication of isolation valve leakage; and, as a result, starting in 1966 the leakage test was performed by pressurizing the steam line between the isolation valve and the turbine stop valves with air. The following is a listing of the results of the tests done in 1966, 1967 and 1968. The numbers are the leakage of the two main steam line isolation valves in terms of percent of the containment (sphere) volume.

1966 .07%/24 hours @ 37 psi

1967 .140%/24 hours @ 37 psi

1968 .108%/24 hours @ 37 psi

sphere volume = 2,876,000 ft^3

The above valves were averaged and one-half of the average leakage was assumed for the leakage of each valve. It was then assumed that this leakage would be the expected leakage for each Dresden 2 isolation valve after these valves have been in service for a comparable period. Corrections for the difference in valve size, and test pressure, were made. This then results in an expected leakage of 75 ft^3/hr at a test pressure of 25 psi or approximately 3% of the contained volume per day.

Supplementary Information - DRESDEN 2 MAIN STEAMLINE CONTAINMENT ISOLATION VALVE SPECIFICATION AND LEAK TEST RESULTS

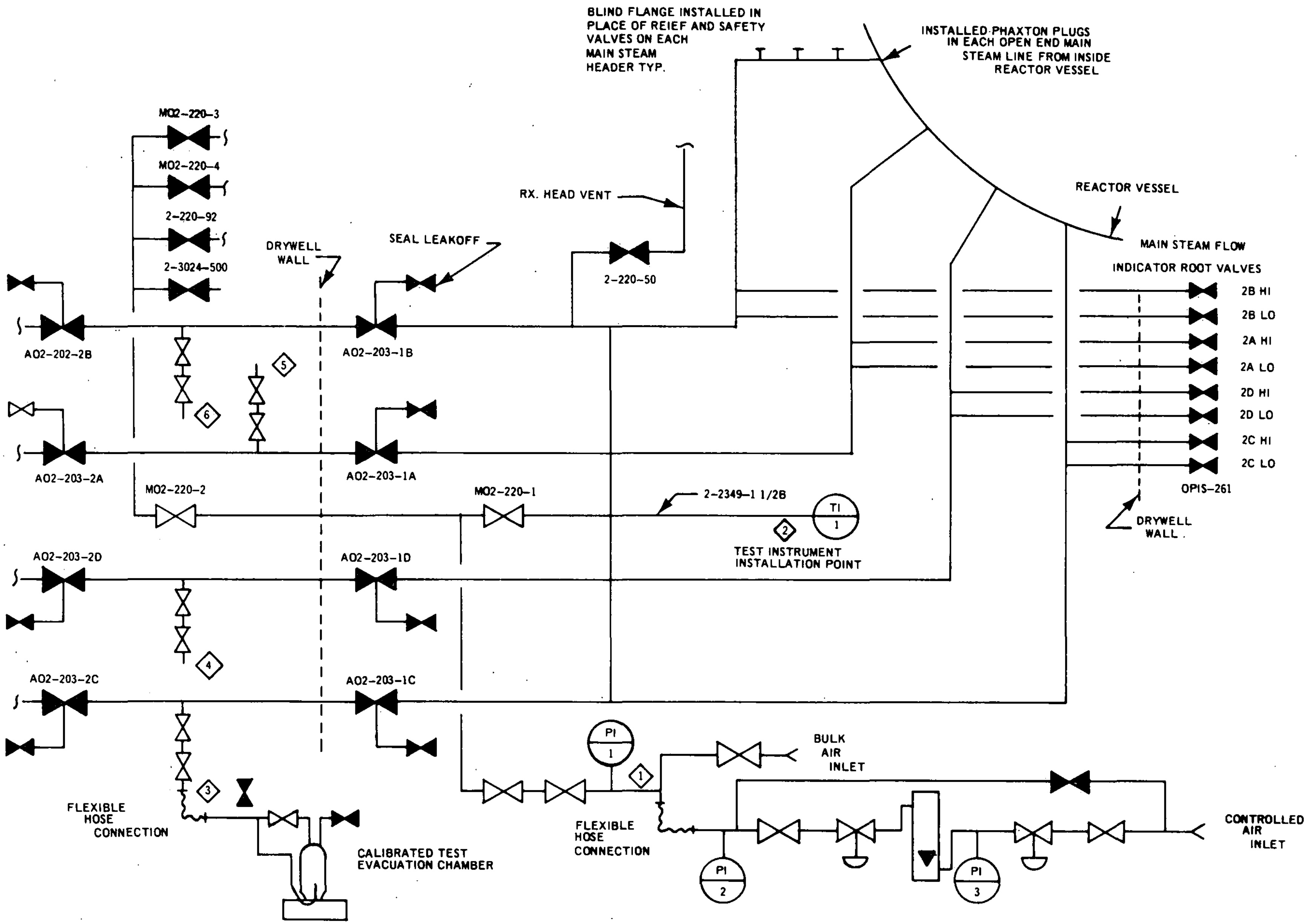
> "Provide a copy of the Dresden 2 main steamline containment isolation valve design specification and the results of the construction leak tests."

During the Unit 2 construction period, leakage tests were conducted on both the inboard and outboard main steam isolation valves. The attached Figure, "Dresden II Main Steam Isolation Leak Testing," outlines the boundaries of system piping placed under test. Filtered, oil free air was used at a sustained pressure of 50 psig and containment volume was calculated at 275,481 ft^3.

Following is a compilation of test results for each valve:

AO-2-203-A1	less than 1 cfh
AO-2-203-A2	less than 1 cfh
AO-2-203-B1	2 to 3 cfh
AO-2-203-B2	less than 1 cfh
AO-2-203-C1	18 to 21 cfh
AO-2-203-C2	18 to 21 cfh
AO-2-203-D1	less than 1 cfh
AO-2-203-D2	less than 1 cfh

Subsequently, valves AO-2-203-C1 and C2 were disassembled and indentations were found in the seats. After lapping the seats of each valve, they were retested. The leak rate found for these valves was 2.5 cfh.



DRESDEN II MAIN STEAM ISOLATION LEAK TESTING

57.16

Supplementary Information - LEAKAGE RATE OF ISOLATION VALVES

> "Establish a reduction factor which can reasonably be shown to exist for leakage from the isolation valve to the environment. Recognize the use of the TID 14844 source term".

An analysis has been performed of the offsite consequences of leakage from the containment via the steamline isolation valves through the steamline and into the condenser. This analysis shows the permissible leakage rate for the isolation valves considering this leakage path. Two methods of analysis are presented; one considers the effects of plateout and partition of iodine to yield realistic but conservative estimates of doses; the other considers very conservative assumptions neglecting all mitigating factors and assuming leakage from the containment is transferred directly to the condenser. In both methods the fission product source term described in TID 14844 is presumed released from the reactor core.

Both analyses methods assumed that the steamline between the containment isolation valves and the turbine is an intact barrier which directs and confines the leakage through the isolation valve to the condenser. This pipeline has received for the most part the same design, fabrication and installation quality assurance as the steamline inside the containment. Thus, there is a high degree of assurance that the integrity of this steamline will be maintained subsequent to an accident to direct isolation valve leakage. A discussion follows to describe the quality of this steamline.

The steamline beyond the containment isolation valves has received high quality design, fabrication and installation such that it meets the criteria of a Class 1 piping system. Specifically, the following design and quality assurance provisions were taken.

1. The piping required 100% radiography of all welded joints in plate pipe and butt-weld fittings with techniques and standards of acceptances complying to ASME Boiler and Pressure Vessel Code, Section VIII, Paragraph UW-51. All circumferential butt welds

made during shop fabrication or field fabrication were similarly radiographed.

2. The pipe was grit-blasted and was shipped to the plant site with open ends metal capped and taped with silica-gel placed inside the pipe to insure against corrosion. Metal protectors were reinstalled on the open ends of all piping during erection at the end of each working day.

3. Supplementary tests S-5 and S-6 were performed as described in ASTM A106. The check analysis required in the supplementary tests were made for each heat of piping involved and is essentially an etching test to insure the integrity of material.

4. 100% radiography was performed on all valve castings in sizes 4" and larger for 900-lb class valves. Radiography on lower pressure class valves was performed on the weld ends and critical casting locations.

5. A seismic analysis was performed on the piping similar to that within the containment. The piping meets the Class 1 seismic criteria.

Considering the above standards applied to the steamline, there is a high degree of assurance that its integrity will be intact subsequent to an accident to direct leakage from the isolation valve to the condenser.

I. Analysis Using AEC Assumptions

Although the actual calculational techniques employed by the AEC are unknown, the following assumptions are known to be used by the AEC.

A. Assumptions

1. Iodine activity released from damaged rods - 50%.
2. Percent of core damaged - 100%.
3. Plateout in primary containment - 50%.
4. Suppression pool effects, washout, etc. - 0.
5. Leakage from primary containment directly to standby gas treatment system (SGTS) or to main steam line isolation valves - 1%/day constant non varying with pressure (total leakage of 2%/day).
6. Height of release - 100 meters for SGTS discharge and zero meters for condenser release.
7. Breathing Rate - 2.47×10^{-4} m^3/sec $o<t<8$ hrs and 2.32×10^{-4} m^3/sec for $t>8$ hrs.
8. Meteorological conditions:

Parameter	Time Interval Appropriate	Parametric Value
Wind Speed	0-1 day	1 m/sec
	> 1 day	2 m/sec
Stability (0m)	0-1 day	Type F (100%)
	> 1 day	Type F (50%), Type D (50%)
Stability (100m)	0-1 day	Type C (100%)
	> 1 day	Type C (50%), Type F (50%)
Wind Blows in 1 direction	0-1 day	100%
	1-4 days	50%
	> 4 days	33%

9. Reflection factor - 2, for both release heights.
10. Building Dilution Effects - Considered for H=0.
11. Filter efficiency - 90%.
12. Condenser leak rate of 0.5%/day continuously.

B. Results

The thyroid inhalation dose at the site boundary (0.5 miles) and the low population zone (5 miles) is shown in Table I for both the isolation valve contribution and the standby gas treatment contribution as well as the total dose. It should be emphasized that in arriving at these exposures the meteorological conditions defined in assumption IA-8 above for the stack and ground level releases assumed to occur simultaneously, which is an impossible situation. As noted in Table I the 2 hour thyroid close at the exclusion area (site boundary) is 18 rem while the 30 day low population zone dose is 76 rem.

II. Psuedo-Realistic Approach

The assumptions stated herein are considered only psuedo-realistic as the conditions specified in assumptions IA excluding items 5, 11 and 12 are also considered valid for this calculation. The leak rate (initially the same as IA-5) item 5, is based upon the actual containment pressure with an average leakage rate for the duration of the accident of 0.21%/day to the condenser assumed applicable. The fission products released to the standby gas treatment system (SGTS) are assumed to be uniformaly mixed in the secondary containment prior to discharge to the SGTS. The assumption is also made that the filter efficiency, item 11, applicable is 99% compared to 90% considered in the previous case. The activity released from the condenser occurs at an average leak rate, item 12, of 0.86%/day with the leakage being mixed in the turbine building prior to release to the environment. In addition a reduction factor of 10 is also allowed for fission product removal in the steam piping and/or condenser due to condensation position and plateout effects.

A. Results

As noted in Table I the 2 hours site boundary thyroid dose is 2.0×10^{-2} rem while the 30 day low population zone dose is 2.3 rem.

III. Conclusions

Based upon the values presented in Table I it can be concluded that with even the most pessimistic assumptions, filtered containment leakage rates of 9%/day (108 rem) and isolation valve leakage rates of 3%/day (192 rem) will not results in exceeding the guidelines setforth in 10CFR100 (100 rem). This 3%/day rate is equivalent to 75 ft^3/hr from each of four valves as discussed on page 4-6.

TABLE I

Radiological Doses As A Consequence of Containment And Isolation Valve Leakage - Thyroid Inhalation Dose

Location	Assumed AEC Approach - Dose* Consequences (Rem) For			Semi-Realistic Approach - * Dose Consequences (Rem) For		
	1	2	3	1	2	3
Exclusion Area (0.5 mi)	$1.7x10^{1}$	$6.9x10^{-1}$	$1.8x10^{1}$	$1.4x10^{-2}$	$5.5x10^{-3}$	$2.0x10^{-2}$
Low Population Zone (5 mi)	$1.2x10^{1}$	$6.4x10^{1}$	$7.6x10^{1}$	$7.3x10^{-2}$	$2.8x10^{0}$	$2.3x10^{0}$

1 - signifies dose contribution from activity released via the stack.

2 - signifies dose contribution from activity released via the isolation valves.