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April 5, 1977

Mr. Dennis L. Ziemann, Chief
Operating Reactors - Branch 2
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Quad-Cities Station Units 1 and 2
Additional Information Concerning
10 CFR Part 50 Appendix J
NRC Docket Nos. 50-237/249/254/265

Reference (a): D. L. Ziemann letter to R. L. Bolger dated February 2, 1977.

Dear Mr. Ziemann:

Reference (a) listed several questions concerning the Dresden Station Units 2 and 3 and Quad-Cities Station Units 1 and 2 Appendix J review. Attached is our response to those questions.

If you require any further information, please contact this office.

One (1) signed original and 57 copies are provided for your use.

Very truly yours,

M.S. Turbak

M. S. Turbak
Nuclear Licensing Administrator
Boiling Water Reactors

Attachment

Additional Information Concerning
10 CFR Part 50 Appendix J

Question 1: Testing of Instrument Lines

References:

1. Dresden and Quad-Cities Submittal to NRC dated September 26, 1975 Regarding Exemptions to Appendix J of 10 CFR Part 50.

2. Letter from D. L. Ziemann to R. L. Bolger dated February 2, 1977 Regarding Additional Information for the Submittal.

3. Section 3.C of Amendment 22 of Dresden FSAR.

4. Response to Question 5.9 Section 5 of Amendment 13 of the Quad-Cities FSAR.

5. Response to Question 5.0 Section 5 of Amendment 23 of the Quad-Cities FSAR.

6. Response to Question 2.0 Section 5 of Amendment 25 of the Quad-Cities FSAR.

The evaluation that demonstrates both stations are in accord with the provisions of Regulatory Guide 1.11 and its supplement can be found in Amendment 22 of the Dresden FSAR and in Amendments 13, 23, and 25 of the Quad-Cities FSAR (see references 3, 4, 5, and 6). In these amendments, a discussion of the 1-inch instrument lines and an analysis of an instrument line break is made. This analysis is summarized in the following paragraphs.

Each of the four units in question contain 16 penetration assemblies which are used for primary system instrumentation. Each assembly is configured to carry six instrument pipes through the biological shield. Each line is a one-inch schedule 80, type 304 stainless steel pipe and is welded to a stainless steel pipe that is welded to the drywell penetration housing. All welds were dye penetrant tested, and the lines meet seismic Class 1 requirements.

Of the total 96 penetrating pipes for each unit, 77 are active lines and 19 are spares. Each of the active lines have stop valves and excess flow check valves located outside the containment. On all but three lines, the process stop valves are no more than one foot from the penetration. The excess flow check valves permit a maximum flow of 2 gpm, and these check valves are checked once per operating cycle.

If one of these lines should break outside the primary containment upstream of the flow check valve, approximately 100,000 pounds (12,000 gallons) of water and steam will be released to the reactor building before the reactor is shut down according to the analysis presented in the FSAR amendments. This amount of coolant loss is well within the capability of the reactor coolant makeup system. The maximum reactor building pressure that is reached following the line break is 0.75" H_2O. A building pressure of 7" H_2O is needed to lift the blow-off panels. Therefore, the integrity of the secondary containment would be maintained, the building filters would not be bypassed, and the standby gas treatment system would be operable. The maximum off-site exposure depends on what assumptions are made, but in the worst case it is 6 rem to the thyroid which is substantially below the guidelines of 10 CFR Part 100. The consequences of any line break connected to the containment atmosphere would be much less.

Question 2: <u>Sequence of Local and Integrated Leak Rate Tests</u>

We believe that compliance is achieved with 10 CFR Part 50 Appendix J in performing local testing prior to the integrated test. According to Section III.A.1.(a) of Appendix J, repairs or adjustments shall be made to components whose leakage exceeds that specified in the Technical Specification, during the period between the completion of one integrated leak rate test and the containment inspection just prior to the subsequent test. The components mentioned relate to seals, penetrations, and valves which are local leak rate tested prior to the integrated test. The local tests performed in this fashion are conducted such that "as-found" conditions exist. All double-gasketed seals are tested for leakage prior to any opening, and all isolation valves are tested after their initial closure during the refueling outage. The requirements of Section III.A.1.(b) of Appendix J are met in that all valves are closed by normal means. If the as-found measured isolation valve leak rate exceeds Technical Specification limits, repairs are made and re-tests are performed. Then, the difference between the as-found and corrected leak rates is added to the subsequent integrated leak rate test results. Additionally, the leak rate from certain isolation valves, which are not exposed to 48 psig containment atmosphere during the integrated leak test, are added to the integrated test results. These aforementioned measures are used in order to obtain a meaningful and correct as-found leak rate value related to the performance of the integrated leak rate test.

The commission states that a single end-of-outage integrated leak rate test is permissible "if the conservative assumption is made that the total measured local leak rate is in a direction out of the containment."

A. In those cases where the combined leakage of two isolation valves is measured in a single test by pressurizing between the valves, the above assumption cannot apply since under accident conditions, the leakage out of the containment via such a penetration would have to pass through the two isolation valves in series. The actual leakage would be equal to the smaller leak rate of the two valves since it effectively throttles the flow through the penetration. In these cases, we intend to make the most conservative assumption possible - the valves leak equally.

B. The assumption that "the total measured leakage (during local leak rate testing) is out of the containment" is not representative of actual containment outleakage for the following reasons:

(1) When the individual local leak rate test results are accumulatively applied to the integrated leak rate test, a multiple single failure criteria is imposed which is unnecessarily conservative.

(2) The regulation does not require this type of "worst possible case" determination, but rather an "as-found" integrated leakage.

(3) Since we are proposing to extrapolate the "as-found" value from a single end-of-outage integrated test plus the local leak rate tests, we recognize the need for conservative assignment of the measured leakage. However, we feel that the assumption that all the boundary valves in the LLRT leak equally should be sufficiently conservative especially considering the additive effect of the individually determined leak rates. Please consider the following example.

a. For a test of the two isolation valves in a single penetration assume an extreme case where one valve fails to shut properly and therefore leaks at 400 SCFH while the other valve operates normally and leaks 2 SCFH.

1. Actual penetration through leakage could not exceed 2 SCFH since one valve is essentially leak tight.

2. The method we propose would establish a total measured leakage of 402 SCFH, resulting in a very conservative estimate of penetration through leakage equal to 201 SCFH.

3. It is not necessary to know the portion of leakage through each valve since no combination of individual leak rates can result in a penetration through leakage greater than the 201 SCFH estimate.

4. Further tests including visual and sonic leak detection are then performed to identify the major leakage paths so that repairs can be made.

Questions 3 and 4: <u>Airlocks and Test Pressure for the Airlock</u>

A. The commission "requires that reactor containment airlocks be tested at peak calculated accident pressure (Pa) at six month intervals" reasoning that airlocks "represent a potentially large leakage path that is more subject to human error than other isolation barriers."

(1) The commission staff has represented Pa as a gauge value equal to the peak accident pressure. Appendix J defines Pa as "the calculated peak containment <u>internal pressure</u> related to the design basis accident," recognizing the directionality of Pa as a positive pressurization of the containment pushing outward against its boundaries. The proposed increase in pressure between the airlock doors would make the test less representative of design basis accident conditions.

(2) The airlock is "designed to seal the door against a pressure of 2 psig and against 62 psig pressure of the containment vessel existing in the vessel or vessel and lock." Were the airlock to be tested at Pa, the inner door and door mechanism would be subjected to a force of approximately 172,000 lbs. <u>in excess of design</u>. Even with the normal mechanism augmented by the use of strongbacks, such a test is inconsistent with good engineering practice and presents an unacceptable safety hazard. In addition, the use of special restraints is contrary to the premise

that meaningful data requires containment boundaries be set without employing extraordinary means.

(3) The stated objective of the six month test interval is to "provide an integrated leakage rate for the entire airlock assembly, including electrical and mechanical penetrations, the airlock cylinder, hinge assemblies, welded connections and other potential leakage paths." These items form parts of rigid boundaries which are not subjected to mechanical cycling, or to the mating of seating surfaces, or to human error. The airlock should, therefore, be pressure tested at the same once-per-cycle interval as other containment penetrations.

B. The commission requires an "after each opening" test as "a means of assuring that the door seals had not been damaged or improperly seated during airlock use." Testing at a pressure which would not require the use of clamping devices is recommended for this case.

(1) Any test of the airlock made by pressurizing between the inner and outer doors, even at reduced pressures, presents several inherent problems. First, even at 1 psig, the nearly two tons of force exerted against the massive inner door would cause the serious threat of equipment damage. Secondly, no practical means of having personnel enter the drywell to inspect the inner door, either during or following a test, is available even if the door could be safely approached. Thirdly, though the test would establish the airlocks' ability to maintain the door's mechanically latched against a large force attempting to lift the inner door off its seat, it would not necessarily be a meaningful representation of its ability to perform its safety function.

(2) In view of the fact that there have been no airlock door seal failures, we propose a detailed visual examination following each series of entries in place of the test at Pa or at reduced pressure, confident that the objectives of the "after each opening" test can be met with comparable reliability and with timely identification of developing problems.

Question 5: Traversing Incore Probe System

During the Dresden 1976 Fall refueling outage on Unit 3, TIP system and purge line valves were successfully tested by disconnecting the TIP tubes at fittings just inside the drywell. This technique will permit routine LLRT's to be performed without modification of the TIP system and will be utilized in future refueling outages. The testing will commence at Quad-Cities with the present Spring 1977 refueling outage on Unit 1.

Question 6: Additional Questions for Dresden Station Only

A. Drywell air sample valves.

(1) During the next refueling outage a method similar to that used on the TIP system will be developed to test all air-operated containment atmosphere sampling valves. Fittings inside the installed Beckman Analyzer will be used to conduct the tests without system modification.

B. Isolation condenser vent valves.

(1) A temporary exemption is required for the isolation condenser vent valves, which vent the two condenser steam inlet lines back to the main steam line thus providing a standby warming feature. The bounded volume required to pressurize against these valves from the reactor vessel side cannot be established because of the existence of an unisolable line to the "A" main steam header. The volume bounded by the two vent valves cannot be pressurized because of the lack of a test connection on the line. The bounded volume required to pressurize against these valves from the isolation condenser side can be established but cannot be pressurized because of the lack of a test connection on that line.

(2) During the next extended refueling outage (following design completion) for each unit, a modification program will be initiated to make the isolation condenser vent valves testable.

C. ECCS injection valves.

(1) All ECCS injection valves are currently testable and no exemption has been sought for these valves.

A possible source of the confusion is the September 9, 1976 letter to Mr. Galler in which we stated:

"the high pressure coolant injection and low pressure coolant injection and core spray suction valves are non-testable. These valves remain open during accident conditions."

If this was interpreted to mean the HPCI injection valve and LPCI injection valve and core spray suction valve, we would like to offer the following clarification:

The exemption is being sought for the HPCI, LPCI, and core spray suction valves.

(2) A temporary exemption is required for the HPCI system isolation valves in the suction line from the torus. Under certain post accident conditions, these valves auto-open to transfer suction from the condensate storage tanks to the torus. The volume bounded by these two valves cannot be drained and therefore is currently non-testable. (Both the test and drain connections for this line are located upstream of check valve 2301-39.) During the next extended refueling outage (following design completion) for each unit, a modification program will be initiated to install a drain connection downstream of check valve 2301-39 allowing local leak rate testing of this penetration.

(3) Unlike the HPCI suction valves described above, core spray and LPCI have individual suction valves in each pump suction line. These valves have no control function, do not operate intermittently, do not respond to any isolation signal, and do not act as post accident isolation valves. The valves are locked open to assure a suction path from the torus and should therefore be exempt from local leak rate testing.

D. Reactor building closed cooling water (RBCCW) supply and return valves.

(1) Inside the containment the RBCCW system is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. Outside the containment, the RBCCW system is neither a part of any

other systems' pressure boundary, nor connected directly to the reactor building atmosphere. Both the supply and return headers contain a single manually-controlled motor-operated isolation valve in accordance with 10 CFR Part 50 Criterion 57. Our request for an exemption for these valves is based on the following:

a. The special "closed loop inside the drywell/closed loop outside the drywell" construction of this system insures its integrity even with a single failure. The worst case accident, a catastrophic pipe failure on the return line just inside the contained area, would eventually allow the containment atmosphere to enter the RBCCW system (after the header had drained back to the drywell), but it would still be contained within the closed loop outside the drywell.

b. The Technical Specifications do not list these valves as "primary containment isolation valves."

c. The FSAR states that isolation valves in lines which form a closed loop, either within the containment or outside the containment, will not be separately leak tested.

d. Extensive system modifications including major valves in the supply and return lines as well as test connections would be required to make this system testable. These modifications would neither improve system safety nor affect containment integrity.

In addition to the above, Dresden and Quad-Cities would like to submit the following exemption request for the feedwater check valves.

Because feedwater check valve local leak rate tests (LLRT's) have repeatedly yielded excessive "as-found" leakage results, a new test method has been developed which approximates as closely as possible the primary containment conditions during a loss-of-coolant accident (LOCA). Though we are confident that the new technique used in testing these check valves satisfies the intent of 10 CFR Part 50, Appendix J, after notifying Region III of the new method (letter to J. G. Keppler from B. B. Stephenson dated December 14, 1976) we received verbal instructions from Region III to make a formal request for an exemption from the test method described in Appendix J.

10 CFR Part 50, Appendix J, Section III.C.1 specified the following:

A. Local pressurization.

B. Pressure applied in the same direction as when the valve would perform its safety function.

C. Closing by normal operation without preliminary exercising or adjustments.

Previously, all tests have been initiated by closing all manual, motor, air, and solenoid valves in accordance with Item C above. But, if a check valve formed part of the test volume boundary, the local pressurization step was relied upon to provide the backpressure/backflow environment which acts as the normal closing operation allowed by Item C.

Our past experience with the feedwater check valves indicates that due to their size and design, and the large volumes being tested, reliance upon the local pressurization to provide the normal closing operation conditions is not adequate.

A new test method was developed based on the FSAR requirement that the feedwater check valves "must be closed to assure containment integrity immediately after a major accident" and the FSAR design "closure signal - reverse flow."

The conditions selected were the design basis LOCA followed by a loss of off-site power, two failures which would result in the lowest differential pressure available to seat the feedwater check valves. For these conditions, reactor and containment pressure would be approximately 50 psig when the feedwater system pressure dropped to zero as the result of the loss of the off-site power. Postulated LOCA's with less than design basis size breaks would result in longer blowdown rates and potentially higher differential pressure available to seat the check valves following the subsequent loss of off-site power. Since there would still be water on the valves due to their position in the low point of the line, a method of seating the valves with water is representative of accident conditions. Therefore, the testing of the feedwater check valves will be preceded by filling the volume upstream of the valve with water and pressurizing it to approximately 50 psig. This step simulates as closely as possible the normal closing operation allowed by Section III.C.1. The test volume is then drained to assure the water does not act as a fluid seal and a LLRT is performed utilizing isolation valve test procedures. Current isolation valve test procedures specify a pneumatic pressure decay test as required by the Technical Specifications.