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50-237

July 16, 1982

Paul O'Connor
Project Manager
Operating Reactors Branch No. 5
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden 2
SEP Topic: III-5.B, Pipe Break Outside Containment

NRC Docket 50-237

Reference:
1) Telecopy dated March 16, 1982 from G. Cwalina (NRC) to S. Powers (CECo).
2) March 21, 1980 letter, R. Janecek (NLA-CECo) to D. Ziemann (NRC).
3) January 17, 1980 letter, D. Ziemann to D. Peoples.

Dear Mr. O'Connor:

In response to Reference (1) requesting additional information, the following information is being provided.

Item 1: We would like to discuss what stress calculations you have available in order to access the significance of differences from the BTP criteria.

Response: Item 1 arises from Reference (2) where a comparison of the design of the containment penetration piping outside containment between the containment and the first outside containment isolation valves for the main steam lines, isolation condenser steam and condensate lines, and reactor water cleanup inlet line with the provisions of section B.1.b of Branch Technical position MEB 3-1 (appended to Standard Review Plan 3.6.2) were addressed for comparison of original part design USAS B31.1-67 code allowable stresses vs. ASME Section III, Class 2.

Attached are the following:

a) Sargent & Lundy Engineers stress analysis for each of the four main steam lines including the analytical model drawings.

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| 1) | Main Steam, MS-A | Drawing M-1221, Revision A (sheet 1 thru 5) |
| 2) | Main Steam, MS-B | Drawing M-1222, Revision A (sheet 1 thru 6) |
| 3) | Main Steam, MS-C | Drawing M-1223, Revision A (sheet 1 thru 6) |
| 4) | Main Steam, MS-D | Drawing M-1224, Revision A (sheet 1 thru 4) |

b) EDS Nuclear Inc. pipe stress data including mathematics model drawings are provided for the Isolation Condenser (ISCO) & Reactor Water Cleanup (RWCU) Systems.

1) Isolation Condenser (ISCO):

Drawing D2-ISCO-02B(C), Revision 2: Isolation Condenser Drain lines

Drawing D2-ISCO-02C, Revision 1: Isolation Condenser Steam Supply (outside drywell)

Drawing D2-ISCO-01B(C), Revision 0: Isolation Condenser System fill lines

Drawing D2-ISCO-01C, Revision 1: Isolation Condenser Piping Condensate return outside drywell

2) Reactor Water Cleanup (RWCU):

Drawing D2-RWCU-01 B/C, Revision 0: Reactor Water Cleanup Piping Lines: 1202-10", 1201A-8", 1201B-8", 1202-8" & 12-3-8"

<u>Item 2</u>: Reference (3) states that ESF cable trays would be affected. Your letter says that ADSA could be used with the CRD pumps for reaching shutdown. Do the ESF cable trays, then, not contain any cabling needed for ADS operation?

<u>Response</u>: More accurately a determination of whether any cable associated with the ADS, Isolation Condenser Steam and condensate isolation valves, or the CRD System are routed through the Feedwater Regulating Area. A feedwater line break in the area could damage these cables.

From Sargent & Lundy Engineers fire protection study for associated circuits, it has been determined that:

a) No CRD pump cables are routed near the Feedwater Regulating Station. However, a postulated break could affect the 4KV tie between switchgear 23-1 and 23. Therefore, CRD Pumps 2A would be unable to receive diesel power.

It is possible to power CRD pump 2B for safe shutdown in the event of a feedwater line break in the feedwater regulation station (FRS) area. However, the following operator actions would be required due to the loss of control cables in the FRS area:

1. Manual start of the Unit 2 diesel, since autostart could be affected by the postulated break.
2. Manual control of the 4KV feed breakers.
3. Manual transfer of Division II 125 VDC control power from the main to reserve supply.

Despite the above actions, a single failure of CRD Pump 2B would preclude the use of the CRD Pumps without a crosstie between the CRD pumps of Unit 2 and Unit 3. This crosstie is proposed in the Fire Protection study on associated circuits.

b) Control cables to the Isolation Condenser isolation valves do pass through the FRS area. The Fire Protection Study on associated circuits postulates that a feedwater line break damaging all cables in the FRS area is considered to result in a spurious signal closing the in-containment Isolation Condenser valves and loss of control of those valves. Since the feedwater break affects only control cables to these valves, the operator could restore the valves to the open position by jumping contacts at the motor control center. However, the valves outside containment can be manually operated.

c) ADS cables also pass through the FRS area, but ADS is not needed to bring the reactor to a safe shutdown condition. However, since the Isolation Condenser can be used, additional depressurization capability is not required. Overpressure protection is the required function in the initial 5-minute period before the isolation condenser can handle the entire decay heat load. Since the ADS is not required, it is of no consequence that ADS cables are in the FRS area. The Target Rock Valve 2-203-3A can open by main steam line pressure and the Safety Relief valves 2-203-4A through 4H are available to provide the necessary overpressure protection.

Item 3: Several fire protection modifications are discussed in Reference (2). Have these changes been implemented?

1. HVAC duct fire damper.
2. HVAC duct 18" curb around the damper.
3. Fire protection water system leak alarm (see p. 5 of reference 3).

Also, has the drain pipe rerouting to radwaste been completed?

Response: As part of the fire protection modifications, a fire damper was installed per Station Modification #M12-2/3-78-5 and an 18" curb was installed around the damper per Station Modification #M1-2/3-78-6, Structural building components.

A fire protection water system leak alarm was installed per Station Modification #M12-2/3-78-4 to improve the operator ability to detect and stop fire system leaks before essential equipment is flooded.

Also, the drain pipe rerouting to radwaste was completed in 1980 so that the discharge of the drains would not go to the river.

Item 4: Is there any equipment on the 517 elevation that is vulnerable to damage from water draining down from the RBCCW leak?

Response: More specifically, would there by any effects on safety-related 480V motor control centers (MCC's) on elevation 517'-6" of the Reactor Building due to an RBCCW line break on elevation 545'-6".

The RBCCW system is a moderate-energy fluid system. According to SRP Section 3.6.2, cracks should be postulated in moderate-energy fluid system piping. The fluid flow from such cracks should be calculated assuming a circular opening in the pipe of an area equal to that of a rectangle with one side equal to one-half the pipe diameter and the other side equal to one-half the pipe thickness. Using these parameters, the flow from a crack in the 24-inch diameter RBCCW Pump discharge header is about 623 GPM.

At least 300 GPM will be collected by floor drains on elevation 545'-6". The remaining will flow through floor openings to elevation 517-6". Nearly all the water reaching elevation 517'-6" will flow into floor drains on that elevation. A negligible amount of water will flow through floor openings on elevation 517'-6" to elevation 476'-6". No accumulation of water is expected on any elevation.

MCC's 28-1, 28-7, 29-1, 29-4, and 29-7 are located on elevation 517'-6". These MCC's are mounted on low pads but are not surrounded by curbing. However, since no water will accumulate on elevation 517'-6", the only possible hazard to an MCC is water falling directly onto the MCC from a floor opening above.

Only MCC 28-1 is under floor openings that could allow water to fall on it. Loss of MCC 28-1 would disable operation of the Isolatin Condenser valves and disable safety systems powered by Division I. However, Division II would be unaffected and safe shutdown could be reached with the Division II HPCI and LPCI Systems.

Please address any questions you may have concerning this matter to this office.

One (1) signed original and forty (40) copies of this transmittal have been provided for your use.

Very truly yours,

Thomas J. Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

SPP/ji
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Attachment
cc: RIII Resident Inspector, Dresden
Gregg Cwalina, SEP Integrated Assessment Project Manager (w/attach.)