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April 30, 1973

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Mr. D. J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Supplement 3 to Dresden Station Special Report No. 14/Supplement 1 to Quad-Cities Station Special Report No. 7, "Response to AEC Questions Concerning the Proposed Containment Atmospheric Dilution for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2", AEC Dkts 50-237, 50-249, 50-254 and 50-265

Dear Mr. Skovholt:

Supplement 3 to Dresden Special Report No. 14/Supplement 1 to Quad-Cities Station Special Report No. 7 contains the responses to the outstanding questions remaining from your letter of November 14, 1972. This letter asked for additional information concerning the proposed containment atmospheric dilution (CAD) system for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.

One signed original and 59 copies of this report are supplied for your use.

Very truly yours,

L. D. Butterfield, Jr.
Nuclear Licensing Administrator

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Received w/Ltr Dated 04-30-73

SUPPLEMENT 3 TO DRESDEN STATION SPECIAL REPORT NO. 14

SUPPLEMENT 1 TO QUAD CITIES STATION SPECIAL REPORT NO. 7

"Response to AEC Questions Concerning the Proposed Containment Atmospheric Dilution System"

AEC Dockets
50-237
50-249
50-254
50-265

Commonwealth Edison Company

April, 1973

QUESTION NO. 1

Discuss experience at Dresden and Quad-Cities Station with respect to the concentration of oxygen you have been able to maintain in containment. In addition, discuss your capability to further reduce oxygen concentration.

ANSWER NO. 1

Dresden Station attempts to maintain the oxygen concentration in about a 2 to 4% range and has been purging at approximately two-week intervals.

Quad-Cities Station maintains the oxygen concentration in about a 3 to 5% range by utilizing approximately weekly purgings.

It is not practical to further reduce the range of oxygen concentration because of the substantial increase in purge time and frequency that would be required. This would require considerable additional venting of the primary containment to the environment, which is not considered desirable.

QUESTION NO. 8

Provide detailed P&I diagrams showing all essential system elements and the detailed design arrangements for the proposed nitrogen dilution and purge backup systems. Include the appropriate sampling, mixing, and makeup system elements.

ANSWER NO. 8

Detailed P&ID's are not available for the Dresden & Quad Cities plants. The criteria details are contained in Special Report No. 14. Detailed design will not be started until AEC approval has been received for the submitted design criteria.

QUESTION 14a

Section 3.2, page 6 of the Supplement to Special Report No. 14, states that "the operator will have sufficient time available to establish some small leakage rate if the containment leakage rate proves to be too small". Clarification of this statement should be provided and should include discussion of:

a. The information readily available to the operator to facilitate his judgment on whether the leak rate is less than or greater than the allowable leak rate.

ANSWER 14a

The CAD System is operated manually. Following a LOCA, records will be kept of hydrogen and oxygen concentrations and pressures in the drywell and suppression chamber, and calculations will be made of the production rates of hydrogen and oxygen in each of these volumes. Nitrogen additions will be made periodically as needed to keep the hydrogen concentration below 4 percent or the oxygen content below 5 percent in each volume. Additions will be made one at a time to the drywell and the suppression chamber. The amount of nitrogen to be added is determined by a simple calculation:

$$V_N = V_i \left(\frac{C_i}{C_f} - 1\right)$$

Where V_N = volume of nitrogen to be added, scf

V_i = volume of gas in drywell or suppression chamber before nitrogen addition, scf

C_i = initial concentration of oxygen or hydrogen

C_f = desired final concentration of oxygen or hydrogen

If hydrogen and oxygen production rates approach those assumed in Safety Guide 7, the containment pressure will increase and may reach the predetermined limit of 30 psig. Before this pressure is reached containment venting will be commenced. Gas releases will be made during periods when meteorological conditions are most favorable. Gas will be released intermittently at a rate of about 100 scfm until the desired volume has been released. Releases are continued until the containment pressure has been reduced to atmospheric.

Three combinations of pressure reduction and time of initiation were examined for containment pressure response. The pressures are shown in Figure 3 of the Dresden Unit 3 Supplement to Special Report 14. Based on this information, venting will be initiated at 10 days in accordance with the pressure curve shown. Approximately 17,300 scf will be released daily. Nitrogen addition will be continued in order to maintain oxygen concentration below 5 percent. This venting procedure will reduce the pressure to atmospheric within about 90 days.

The operator will have available to him information on the pressure, hydrogen content, oxygen content, radioactivity in the containment atmosphere, and amount of nitrogen added for both the drywell and suppression chamber. Meterological information will be available also. Using all of this information an operator can safely follow the venting procedure without exceeding the 10 CFR 100 limits following a LOCA.

QUESTION 14b

The system used and actions that can be taken to control the leak rate.

ANSWER 14b

The system used to release gas from the containment is shown in the FSAR and new detailed design drawing will be submitted later. The existing containment venting system, shown in FSAR is designed to meet seismic Class I requirements. Each nitrogen supply line will have a tap to supply the valves in its flow path (including the gas release valves). The nitrogen-operated valves are controlled from the main control room. Each valve of a given pair receives its power supply from a separate bus. The pressure control valves in the gas release lines reduce the pressure to about 1 psig. Globe valves in the lines are set during preoperational testing to pass a flow rate of about 100 scfm. These settings are checked at each refueling outage. The flow through the vent line, therefore, is substantially constant. The rate at which containment pressure is reduced is varied by varying venting time and/or frequency.

QUESTION 14c

A discussion on what constraints and system provisions will exist for the operator to limit the allowable leak rate within prescribed values.

ANSWER 14c

The gas release rate is limited to 100 scfm, as stated in the response to Question 14a. The operator manually controls venting time and frequency.

ANSWER 14c

Changes in containment pressure are slow. To reduce containment pressure by 1 psi, for example, 19,000 scf of gas would be released over a period of 190 minutes. The gas release is started and stopped by the operator. Being coordinated with meteorological information, the operation is closely supervised, and automatic termination is not considered to be necessary.

QUESTION 14d

What disposition and processing provisions are provided for the discharged gases?

ANSWER 14d

Gases are passed through the standby gas treatment system and released from the 310 foot stack.

QUESTION 15

Discuss the potential for stratification of hydrogen leakage from the drywell into the reactor building or compartments. Discuss the need for positive mixing of the atmosphere in the reactor building or compartments to prevent the formation of localized combustible gas mixtures.

ANSWER 15

The total rate of leakage from the primary containment, based on the design criteria of 2 percent of the drywell volume at 48 psig, is about 16.8 scfm.

The maximum concentration of hydrogen in the containment atmosphere occurs about 2 days after the loss-of-coolant accident and is 13.2 volume percent.

ANSWER 15

All enclosed areas and compartments of the reactor building have been examined with respect to possible hazards resulting from leakage from the containment. This examination showed that leakage to open areas of the reactor building will not produce hazardous gas mixtures. Adequate mixing occurs as a result of the high diffusion rate of hydrogen and some convection. Leakage into the space between the drywell and the surrounding concrete will not present a hazard because there is no ignition source. In addition, there is no ignition source in the space above the drywell head. Similarly, there are no ignition sources in the clearance spaces between the shield plugs for the equipment access locks, and the biological shield around the drywell.

Two areas requiring consideration are the suppression chamber room and the personnel access room. Of the two areas the personnel access room is the worst case. The suppression chamber room has several openings that will allow convective mixing with other reactor building areas.

It is expected that a substantial portion of the total drywell leakage may occur through the two equipment access locks and the personnel air lock. The personnel air lock is 7 ft. in diameter and is equipped with double O-ring seals. Each equipment lock is 12 ft. 10 in. in diameter. It is conservatively assumed that the rate of in-leakage to the personnel access room is 1/6 of the total leakage from the drywell, or 2.8 scfm.

Assuming a leakage rate of 2.8 scfm containing 13.2 volume percent hydrogen, it is calculated that a purge rate of less than 10 scfm is sufficient to maintain the concentration of hydrogen in the personnel access room at 4 percent by volume or less.

A convection flow path exists from the suppression chamber area to the personnel access room through annular spaces around piping and from the personnel access room to the area above through floor sleeves. The return path for the cooler reactor building air is through the stairwell to the lower reactor building area. Calculations show that a differential temperature of 20°F will produce a flow of 10 scfm through the convective path described.

ANSWER 15

At the time of maximum hydrogen concentration in the drywell the temperature difference between the suppression chamber room and the reactor building area above the personnel access room is expected to be 40 to 60°F. After the maximum concentration of hydrogen is reached, the concentration of hydrogen in the containment atmosphere gradually decreases to about 10 percent by volume. As the concentration of hydrogen decreases the required purge flow decreases. The minimum temperature differential will be 15 to 20°F. The same considerations apply to the suppression chamber room.

It can be concluded from this analysis that no hazard is produced as a result of leakage from the containment to enclosed areas of the reactor building. Further, the SGTS will remove the hydrogen from the reactor building so that the hydrogen concentration in the building will not reach a hazardous level. The building volume will be changed at least once every day.

QUESTION 16

For the long-term period following the DBA, discuss the potential degradation of valve structure and penetrations within the primary containment in connection with the capability of the containment and containment systems to maintain (a) structural integrity and (b) required leak-tightness requirements needed during the long term following a loss-of-coolant accident.

ANSWER 16

It is not anticipated that the integrity of the primary containment will deteriorate to a point where excessive leakage will occur following a design basis accident.

The containment shell, electrical penetrations, and piping penetrations are metallic components (with a ceramic filler in the electrical penetrations) that are designed to pressure vessel standards and thus, no degradation will

will occur from temperature, pressure, or radiation damage.

Some of the valves use Nordel and Silicone rubber as the elastomer and seat material and are located outside the concrete shield. Thus, the temperature (continuous $<$ 250°F and exposure dose $< 10^7$ rads) for these locations are less than the service rating for this material. The temperature this material is exposed to could approach 340°F for about 45 minutes and then drop to less than 250°F for the remainder of the accident. Silicone rubber is good for this temperature range (up to 340°F maximum). The exposure dose after 12 hours is 10^7 rads which is approaching the radiation damage limit for this material. However, the valves and valve seats have already served their function by this time (12 hours); that is, they have prevented bypassing of steam and thus, pressure suppression has been assured. Leakage after pressure suppression has occurred provides additional confidence that the drywell and wetwell are at the same pressure.

The manway into the suppression chamber, the two equipment access locks, the personnel access lock, and the drywell head all have double O-ring seals. These O-rings are made of Nordel which has a continuous temperature rating of 250°F and a 10 hour rating of 340°F.

The temperature of the primary containment walls has been shown to be a maximum of 320°F. The time above 250°F will be less than 10 hours; therefore, temperature will not have an effect on these O-rings. The radiation damage limit is greater than 5×10^8 rads whereas the maximum calculated exposure doses are less than 5×10^8 rads (based on TID 14844 releases) at 100 days. Thus, we have adequate time before the radiation damage limit is reached to reduce the containment pressure to atmospheric as discussed in Question 14a above.

ANSWER 16

All other isolation valves in the primary containment system utilizes metal seats, and therefore, the structural integrity and leak-tightness of these valves will remain essentially unchanged following a DBA.

Buna-N rubber, Teflon, and nylon are used in certain applications in the valves discussed above, but these materials are used only in such locations that their failure will not alter the structural integrity operability of these valves.