Consumers
Power
Company

Regulatory Docket File

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Director of Nuclear Reactor Regulation
Att: Mr Don K Davis, Acting Branch Chief
Operating Reactor Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155, LICENSE DPR-6 - BIG ROCK POINT PLANT - FURTHER DATA ON POSTULATED REFUELING ACCIDENT INSIDE CONTAINMENT

By letter dated April 29, 1977, Consumers Power Company was requested to provide additional information concerning our submittal on the potential radiological consequences of a postulated fuel handling accident inside containment at Big Rock Point, dated March 21, 1977. The purpose of this letter is to provide the necessary response. Additionally, subsequent to the original submittal, minor changes have been made to the location of one radiation monitor and to alarm set points. Therefore, our original evaluation with appropriate corrections is being forwarded as Attachment 1. Further, the information provided in this submittal is based on system design and installation as it will be prior to the 1977 refueling operations. Not all modifications are presently complete, but all changes will be as indicated prior to refueling.

Item 1

Provide the basis for your conclusion that the consequences of the fuel handling accident inside containment are well within the guidelines of 10 CFR Part 100 when reliance is placed on the proposed automatic isolation of the ventilation system. Justify your model for released gaseous radioactivity mixing within containment and containment isolation before the radioactivity is completely released to the environment.

a. Describe the operation of the containment ventilation automatic isolation system and provide schematics of the provisions for automatic isolation of the containment ventilation system. We understand that the automatic isolation of the containment ventilation system will be operative for the June 1977 and subsequent refuelings.

b. Justify the volume of containment air in which the gaseous radioactivity released from a failed fuel assembly is assumed to be mixed before release from the containment.

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c. Indicate the specific ventilation equipment required to be in service during refueling that effects the mixing of the gaseous radioactivity inside the containment.

d. Provide the exact location of the two area monitors which will respond to the accident. Provide a schematic showing the surface area of the refueling cavity that each area monitor would be exposed to.

e. Estimate the time lapse between the containment area monitors' response to the radioactivity from the damaged fuel assembly and radioactivity at the purge line inboard isolation valve. Explain in detail the reasons for the range of response times for each area monitor listed in your response of March 21, 1977.

f. Provide the time lapse between receipt of the containment isolation signal and complete closure of the containment purge line valves.

g. Provide arrangement drawings showing the relative location of the equipment listed in Questions 1.a, 1.c and 1.d above.

Response 1

a. The containment ventilation supply and exhaust valves are of the "fail-safe" design such that they will automatically shut on loss of electrical power or control air pressure. Under operating conditions, they will shut on a high radiation signal from either of the two refueling deck area monitors. On a high radiation alarm, a high impedance relay contact (RS8179 or RS8180) opens, de-energizing coil SVX5. As a result, contact SVX5 opens and coils SVX1 and SVX3 are de-energized venting SV-9151, -9152, -9153 and -9154 and, therefore, closing the supply and exhaust vent valves. The applicable schematics are enclosed as Attachment 2. This sytem will be in operation for 1977 and subsequent refueling operations.

b. As specified in the March 21, 1977 submittal, two extremes of volume mixing were analyzed. These extremes were: (a) No mixing (this assumed a point source dose rate calculation for monitor response) and (b) total mixing (this assumes a semi-infinite cloud dose rate). It is rather obvious that the actual mixing that would occur for these postulated accidents would be somewhere between the two extremes. Howev r, these extrema were chosen to indicate the worst case conditions conceivable and, therefore, bracket any other conditions. Since the resultant "worst case" doses are well within the guidelines specified in 10 CFR 100, we conclude that no further analysis is necessary.

c. The March 21, 1977 evaluation indicated that mixing is not required to ensure that for a postulated refueling accident inside containment off-site doses remain well below the criteria established in 10 CFR 100. Consequently, there are no requirements for the operability of ventilation equipment specifically for this purpose, and none are intended. However, there is ventilation equipment normally in service during refueling. This equipment, its location and description, is listed in Attachment 3.

d. One radiation monitor is located on the west wall of the instrument room at a position approximately 7.6 meters northwest of the reactor center line at an elevation of 635 feet 6 inches. The second radiation monitor is located at the southwest corner of the spent fuel storage pool, approximately 6 meters south-southeast of the reactor center line, at an elevation of 639 feet. These locations are indicated in the schematic forwarded as Attachment 4.

e. Complete automatic isolation of the ventilation system will occur six seconds after a radiation alarm set point is reached. The off-site dose was calculated by combining the six-second actuation time with the applicable time necessary to achieve an alarm set point condition. In order to completely bracket the "worst case" situations, four separate sets of conditions were postulated. These are:

   (1) No Mixing - Peaking Factor = 1.5, Core Decay = 12 Hours
   (2) No [illegible] - Peaking Factor = 0.6, Core Decay = 5 Days
   (3) F[illegible]ing - Peaking Factor = 1.5, Core Decay = 12 Hours
   (4) Full Mixing - Peaking Factor = 0.6, Core Decay = 5 Days

   As a result of utilizing these four sets of conditions, four sets of response times were determined. Since the actual mixing for this postulated accident would be expected to occur somewhere between the extremes of 0 and 100%, the actual response times would, therefore, be bracketed between the calculated extremes.

f. Six seconds as stipulated in our March 21 submittal.

g. Appropriate schematics are provided as Attachment 4.

Item 2

Based on the above information, the source term parameters of Regulatory Guide 1.25, and the assumption that dropping the fuel transfer cask on the core would damage one-third of the core (SER dated June 1962), estimate the offsite doses assuming a postulated worst single failure as requested in our January 17, 1977 letter. For the equipment required to reduce the consequences of this accident, including the area monitors, provide the safety class, power source and technical specification requirements. There should be no reliance on non-safety grade equipment to reduce exposures below the guidelines of 10 CFR Part 100.

Response 2

Since the ventilation isolation valves are fail shut in design, any loss of electrical power or control air pressure will isolate containment. Further, since there are two supply valves in series and two exhaust valves in series, any single failure of a valve will still ensure full containment isolation. The containment ventilation supply and exhaust vent valves receive a "shut" signal from either area radiation monitor; therefore, a failure of one monitor will still ensure isolation. However, assuming that the single failure was the failure of a monitor, the worst postulated failure would be the failure of the closest radiation monitor. This postulated failure was fully analyzed in our submittal of March 21, 1977. As indicated previously, there have been minor changes in monitor location and alarm set point. Thus, the analysis required slight revision and is included in the Conclusion section of Case I - Fuel Transfer Cask Drop of Attachment 1 to this submittal. To summarize, such a failure would delay valve closure by a maximum of one second during which time a thyroid dose commitment rate of 0.06 rad/second would be accumulated by an individual at the site boundary. This would result in an additional dose commitment of 0.06 rad to the thyroid. The resultant total whole body dose at a rate of 0.017 mrad/second would be 0.017 mrad.

The current Big Rock Point Technical Specifications allow refueling operations to continue as long as one of the two area monitors is operational (ref: Technical Specifications Section 6.4.3.(3)). This condition could conceivably lead to the remote possibility of having the single operational radiation monitor fail at the same time the postulated refueling accident occurs. Since there would still be 14 minutes available for the operator to isolate containment prior to exceeding 10 CFR 100 limits at the site boundary under the worst postulated conditions, and since a postulated accident of this magnitude would be readily apparent to all cognizant personnel allowing ample time for reaction, Consumers Power Company concludes that no further restrictions on refueling operations are necessary.

The safety class and power sources for all applicable equipment are listed in Attachment 5.

Item 3

Propose any additional technical specifications needed to ensure that conditions described in items 1 and 2 will be maintained (in a conservative sense) during all fuel handling operations within the containment.

Response 3

Big Rock Point is in the process of converting its Technical Specifications into the current standards. This conversion will incorporate the applicable and necessary limiting conditions for operation, action statements and surveillance requirements that pertain to these systems. It is fully anticipated that this conversion will be complete and implemented by the 1978 refueling outage. Thus, no Technical Specifications changes are contemplated at this time.

David A Bixel
Nuclear Licensing Administrator

CC: JGKeppler, USNRC