14.9 DOSE SENSITIVITY EVALUATION USING ASSUMPTIONS OF THE AEC/DRL (INCORPORATED WITH TID 14844)

The dose sensitivity analysis utilizes the original evaluations of the radiological consequences of the four design basis accidents. The original sensitivity analysis which provides the effects of variation of design parameters on accident doses remains useful and is retained as background material.

14.9.1 Loss-of-Coolant Accident (183 meter release height)

1. The reactor has operated for an extended period at 3440 MWt.

2. 100 percent of the noble gases in the reactor and 25 percent of the iodine instantaneously become available for leakage from the primary containment as an aerosol based on TID 14844.

3. The primary containment volume leaks at a rate of 0.635 percent per day for 30 days.

4. The escaping aerosol immediately flows through the standby gas treatment system and the stack without mixing in the secondary containment building.

5. 90 percent of the iodine entering the standby gas treatment is retained by charcoal filters.

6. Meteorology - For the exclusion area calculations, the concentrations are those at the plume centerline with Pasquill C conditions. For the low population zone calculations, the concentrations are those for Pasquill C, 1 m/sec wind speed for the first day. For the remaining 29 days the conditions are 50 percent Pasquill C, 3 m/sec wind speed and 50 percent Pasquill F, 2 m/sec wind speed. During the first eight hours, the concentrations are at the plume centerline. During the 8-24 hour period, the plume stays within a 22.5° sector. For the 1-4 day period, the plume stays within a 22.5° sector 1/2 of the three days. The plume stays within the 22.5° sector 1/3 of the remaining 26 days.

7. There is a ground reflection factor of 2 for the plume and there is no ground deposition or rain wash out of the plume.

8. The breathing rate is 347 cc/sec for the first 8 hours and 232 cc/sec thereafter.

14.9.2 Refueling Accident (183 meter release height)

1. Assumptions "1", "4", "5", "7", and "8" of the loss-of-coolant accident.

2. Each damaged fuel rod contains 50 percent more activity than the average fuel rod in the core.

3. 20 percent of the noble gases and 10 percent of the iodine contained within the damaged rods are released within two hours.

4. 90 percent of the iodine released from the rods is retained by the refueling pool water.

5. One bundle is assumed to be damaged (49 rods). Appendix N and NEDE-24011-P-A contain or reference current fuel design information.

6. Meteorology - For the duration of the accident the concentrations are those at the plume centerline during Pasquill C, 1 m/sec wind speed.

14.9.3 Steam Line Break Accident (ground level release)

1. Assumptions "1", "7", and "8", for the loss-of-coolant accident.

2. The concentration of radionuclides in the reactor water are those associated with the maximum stack gas release limit which may be proposed as an operating limit.

3. The total mass of steam and water released from the steamline contain concentrations of radionuclides identical with those in the reactor water.

4. All of the radionuclides contained in the steam and water mass released from the steamline are released to the atmosphere at ground level.

5. It is assumed that there is no thermal rise of the steam cloud.

6. Meteorology - For the duration of the accident, the concentrations are those for Pasquill F, 1 m/sec wind speed. Building wake effects are accounted for with the plume shape factors equal to 0.5 and the building cross- sectional area equal to 2660 square meters.

14.9.4 Control Rod Drop Accident (ground level release)

1. Assumptions "1", "6", "7", and "8" for the loss-of-coolant accident.

2. The damaged fuel rods are from the highest burnup (activity) regions of core.

3. 100 percent of the noble gases and 50 percent of the iodine are released from the damaged rods.

4. 90 percent of the iodine released from the damaged fuel rods is retained by the reactor water.

5. The radionuclides released from the reactor water travel to the condenser where 50 percent of the iodine plateout.

6. The leak rate from the condenser is 0.5 percent per day. (Only if mechanical vacuum pump is not operating.)

7. The accident duration is 24 hours.

8. Meteorology - For the duration of the accident, the concentrations are those for Pasquill F, 1 m/sec wind speed. For the exclusion area calculations, the concentrations are those at the plume centerline. For the low population zone calculations, the concentrations are those at the plume centerline for the first 8 hours. For the remaining 16 hours, the plume stays within a 22.5° sector.

14.9.5 Radiological Consequences

Radiological consequences of the design basis accidents have been evaluated in Chapter 14 using the General Electric method of analysis as described in a topical report APED 5756 "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor" (March 1969). Table 14.9-1 contains the doses calculated using the assumptions listed in paragraphs 14.9.1 through 14.9.4 above.

To demonstrate the effects of the various factors involved in making radiological dose calculations, Table 14.9-2 lists many assumptions and their effect of the resulting dose. Note that many of these factors are nonlinear in nature and, therefore, cannot be interpolated or extrapolated without performing sophisticated calculations.

14.9.6 Discussion of Assumptions

The loss-of-coolant accident has generally been interpreted as a complete core melt (10 CFR 100.11(a) Note 1) without consideration of the geometry aspects of molten fuel and its resultant consequences. Only the fission product release has been considered. Such a situation would only be evaluated in light of little or no core cooling protection. It states in 10 CFR 100.10 that ". . . the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

a. Characteristics of reactor design and proposed operation including:

    (1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials,

    (2) The extent to which generally accepted engineering standards are applied to the design of the reactor,

    (3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials, and

    (4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive materials to the environment can occur.

The plant is designed to keep the thermal response of the core below a clad temperature of 2200°F. Because of this the use of TID 14844 assumptions of core melt fission product release do not apply for the plant.

The above referenced topical report (APED 5756) summarizes the technical basis for all assumptions and models used on current generation GE Boiling Water Reactors. The use of this topical report in evaluating the radiological aspects of the plant is consistent with good engineering and actual design.