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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Proposed Operating License Change Request No. 229;
Conference Call Response Documentation

This letter documents our response to concerns the NRC staff identified during conference calls on March 1, 1996, and March 4, 1996. This information is provided in support of the Alternate Steam Generator Tube Plugging Technical Specification change.

Enclosure 1 provides the NRC concerns related to the radiological analyses followed by our response as discussed during the March 1, 1996, conference call. In addition, Enclosure 2 provides clarification of our March 1, 1996, response concerning rotating pancake coil no detectable degradation (RPC NDD) indications left in service. This discussion is provided in response to a conference call with the NRC staff on March 4, 1996.

If you have any questions regarding this submittal, please contact Mr. S. F. LaVie at (412) 393-5856.

Sincerely,

Sushil C. Jain

c: Mr. L. W. Rossbach, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. D. S. Brinkman, Sr. Project Manager

ENCLOSURE 1

Beaver Valley Power Station, Unit No. 1
PROPOSED OPERATING LICENSE CHANGE REQUEST 229;
MARCH 1, 1996, CONFERENCE CALL DOCUMENTATION

QUESTION 1

There are conflicting estimates of the duration and quantities of steam released from the faulted and intact generators to the environment in the following documents: (1) WCAP-14122 and the (2) Duquesne Light Company (DLC) January 4, 1996, submittal.

Based on the remaining text in this question and the discussions held on March 1, 1996, DLC understands the issues to be:

a. It appears that the offsite dose calculation was performed for only two hours. Why isn't the 2-8 hour dose due to plant cooldown via steam dump considered?

b. The offsite dose calculation and the control room habitability dose calculation use different assumptions regarding iodine partitioning.

c. The offsite dose calculation and the control room habitability dose calculations use different assumptions regarding steam releases.

RESPONSE 1

a. The offsite dose calculation was performed to determine the maximum allowable leakage such that the Standard Review Plan limit of 10% of the 10 CFR 100 0-2 hour exclusion area boundary (EAB) would not be exceeded. Given the large difference between the atmospheric dispersion coefficients (χ/Q) for EAB and the low population zone (for which the 2-8 hour doses would be calculated) the 0-2 hour EAB dose is limiting. For this reason, the offsite dose analysis did not assess the additional 2-8 hour release.

b. Both the offsite and control room habitability analyses assumed no iodine partitioning for the faulted steam generator. The offsite dose calculation conservatively assumed no iodine partitioning for the intact steam generators. This assumption was made in the interest of simplifying the analysis and since the releases from the intact steam generators are not major contributors to the overall 0-2 hour dose. The control room habitability analysis assumed no partitioning in the intact steam generators for the first hour, after which iodine partitioning was credited. The one hour period was conservatively set at twice the maximum time estimated to be required to re-establish level control in the intact steam generators.

c. The steam release data specified in the documentation of the offsite and control room analyses were obtained from different sources, but differ only by a few percent of each other. However, with regard to the faulted steam generators, the offsite dose and control room habitability analyses based the release rate on the primary-to-secondary leak rate, effectively treating the steam generator as a pipe. The steam release data were not used.

The release of the pre-event liquid inventory in the faulted S/G was modeled as 0.999999 of the activity being released in 30 minutes while the pre-event steam release was modeled as 0.999999 of the activity being released in one second. For the 0-2 hour release from the intact steam generators, the offsite analysis assumed the technical specification primary-to-secondary leak rate, effectively treating the steam generator as a pipe. The control room habitability analysis assumed the technical specification primary-to-secondary leak rate and the steam releases identified in the analysis. Holdup was modeled in the intact steam generators.

QUESTION 2.a

Regarding the control room inleakage, paragraph 2 on page 9.13-6 of the UFSAR states that the control room is expected to leak at a rate of 156 cfm based on studies con'ucted by Atomics International. In contrast, DLC's submittal assumed a 10 cfm inleakage.

RESPONSE 2.a

The reference to UFSAR page 9.13-6 is correct. However, that discussion refers to the leak rate while the control room is pressurized -- it is out leakage. This particular UFSAR discussion addresses the adequacy of the air bottle capacity to maintain the requisite pressure. In the control room analysis, the total control room exfiltration is modeled as 700 cfm. This is based on the 690 cfm air bottle bleed (established by testing as needed to maintain 1/8" water gage) plus an assumption of 10 cfm for air exchange through access doors while the control room is pressurized. This 10 cfm infiltration was conservatively assumed even though the Standard Review Plan does not specify applicability for the bottled air control room design type at Beaver Valley.

QUESTION 2.b

It is not clear what value should be assigned to the filtered intake flow rate. According to information contained in Figure 9.13-2 of the UFSAR, the filtered air intake flow rate is 400 cfm. However, Technical Specification 3/4.7.7 indicates that the system operates between 800-1000 cfm during operation. A different flow rate of 700 cfm was assumed by DLC in its January 4, 1996, submittal.

RESPONSE 2.b

Figure 9.13-2 is in error. The 400 cfm value applied prior to upgrades associated with Unit 2 licensing. This figure will be updated. The control room analysis conservatively assumed an emergency intake pressurization flow rate of 700 cfm. This is equivalent to the assumed exfiltration rate described above. During the Unit 2 licensing effort, sensitivity analyses established that the higher flow rate actually resulted in lower doses due to a "purge effect."

QUESTION 3

For the main steam line break (MSLB) accident analysis it is not clear from the available information how the bottled air supply and emergency filtration systems are activated. Additional information should be provided on the assigned sensitivity for the control room monitor.

RESPONSE 3

The control room analysis assumed manual isolation of the control room by operator action at T=30 minutes. The Unit 1 emergency operating procedure E-2, Faulted Steam Generator Isolation, has, as the very first step, the requirement that the operator verify control room emergency bottled air pressurization system (CREBAPS) actuation. The procedure step "response not obtained" requires manual actuation.

The operators have several rapid indications of a MSLB. The break assumed here will result in a steam line isolation, safety injection, and a reactor trip. Entry to the EOPs (E-0) occurs on a reactor trip or safety injection. The transition from E-0 to E-2 occurs after the status of the trip, safety injection, electrical power and other safety equipment operability is verified.

QUESTION 4

According to information presented in DLC's January 4, 1996, submittal, the control room monitor has a setpoint of 0.47 mR/hr, but the process safety limit is 1 mR/hr. The reported difference between the two values is due to corrections for instrument uncertainty. It is not clear from this explanation which of the two exposure rates is the assigned sensitivity for the control room monitor. Additional information should be provided on the assigned sensitivity for the control room monitor.

RESPONSE 4

The process safety limit was established as part of the Unit 2 licensing process. Analyses performed at that time indicated that for the most restrictive accident, the 30 day thyroid dose criterion would not be exceeded if a certain airborne concentration was not exceeded. The postulated external dose rate from this airborne concentration was shown to equal 1.0 mrem gamma.

In determining whether or not the monitor will alarm, the accident case is run with normal control room intake and exfiltration (500 cfm unfiltered) the dose rate versus time is plotted. The point at which the projected dose rate exceeds 1 mR/hr is taken as the CREBAPS actuation time. There are, however, postulated delays. For example, a loss of offsite power is projected to occur just as the dose rate reaches 1 mR/hr. For this reason, an actuation delay of 3.25 minutes is added to the time at which the concentration exceeds 1 mR/hr.

Because of instrument uncertainties, the actual setpoint of the monitor is established at 0.47 mR/hr. Thus, the actual isolation may occur as early as 0.47 mR/hr, but no later than 1 mR/hr, depending on the actual summation of uncertainties.

Although, utilizing the average confirmation rate of 51% for RPC NDD indications left in service for postulated leakage calculations provides some conservatism for indications above 1 volt, as indicated by the Beaver Valley Unit 1 data, DLC will apply the larger of 0.7 or the value obtained using the data from the last two inspections as the fraction of RPC NDD included in the beginning of cycle distribution. This additional conservatism will account for potential variations in confirmation rate in successive outages.