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BECo 85-063

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

License DPR-35
Docket 50-293

Request for Information and Status of
Containment Purging and Venting

By telephone on February 3, 1984, and by letter of April 11, 1984, NRC requested additional information to aid in the completion of the containment purge and vent (P&V) review for Pilgrim Nuclear Power Station (PNPS).

Most of the information was provided by Boston Edison (BECo) in a submittal dated April 6, 1984 which responded to the February 3, 1984 telephone request. This submittal is provided to address the remaining requests of April 11, 1984.

Item 1.c.

"Part 2 of your response (BECo response dated February 15, 1983) states that fully qualified valves would not necessitate a 90 hour Technical Specification limit. We (NRC) recognize that the 90-hour per year limit on purging through 20-inch valves may not be appropriate for fully-qualified 8-inch valves if these smaller valves can be closed after a DBA-LOCA in time to prevent the resulting offsite dose from exceeding 10 CFR Part 100 guidelines. However, as stated in Enclosure 3 of our July 30, 1982 letter, purging/venting should be minimized during reactor operation because the plant is inherently safer with closed purge/vent valves than with open lines which require valve action to provide containment. Therefore, we request that you provide, with the amount of radioactivity released, the results of the LOCA dose analysis mentioned in Part 3 of your response. You should also establish a goal which represents a limit on the annual hours of purging expected through each particular valve, as indicated by Item 2.c.i of Enclosure 3 to our letter of July 30, 1982."

Response

The LOCA dose analysis mentioned in our February 15, 1983 letter gave the following 2 hour Site Boundary Dose (328 meters north) based on conservative assumptions: Thyroid: 40.6 REM; Whole Body Gamma: 1.34 (-2) REM; Beta and Gamma Skin: 5.80 (-2) REM.

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The assumptions and data used to support this aspect of the LOCA dose analysis prior to drywell isolation are as follows:

- A large pipe break inside containment instantaneously releases the entire primary coolant mass (457872. lb) to the drywell;

- No core damage or fuel perforation is postulated prior to drywell isolation (t = 17 sec);

- The entire primary coolant inventory of noble gas and iodine, spiked to the current technical specification limit of 20. microcuries/gram, is released to the drywell and is assumed to become airborne, with no credit taken for plateout;

- The curie inventory airborne within the drywell atmosphere free volume of 147,000. ft^3 is:

Isotope	Curies	Isotope	Curies
I-131	7.30 (+2)	Kr-88	3.7 (+0)
I-132	5.97 (+2)	Kr-89	3.7 (-2)
I-133	3.31 (+2)	Xe-131m	3.1 (-3)
I-134	1.66 (+3)	Xe-133m	5.8 (-2)
I-135	8.44 (+2)	Xe-133	1.7 (+0)
Kr-83m	6.0 (-1)	Xe-135m	1.4 (+0)
Kr-85m	1.2 (+0)	Xe-135	4.5 (+0)
Kr-85	4.1 (-3)	Xe-137	1.4 (-1)
Kr 87	3.1 (+0)	Xe-138	4.3 (+0)

- The Standby Gas Treatment System is assumed inoperable following the large break inside containment, due to failure of its associated duct work;

- The release of drywell atmosphere activity through the two 20" purge/vent lines, which are open during de-inerting operations, is via the reactor building vent, yielding an unfiltered ground level release to the environment prior to drywell isolation;

- No credit is taken for mixing within the reactor building secondary containment;

- A constant release rate of 1.61×10^5 lb/hr directly to the environment was assumed for each of the two 20" open lines prior to isolation of the drywell;

- A mixture specific volume of 10. ft^3/lb was assumed for airborne material being released from the drywell;

- The total amount of radioactivity released to the environment, prior to drywell isolation, is unfiltered and is 418. Curies;

- One-hour accident atmospheric dispersion factors calculated for the reactor building vent ground level release were used to determine the 2-Hour site boundary doses (328 meters north);

$$X/Q_{conc} = 9.44\ (-4)\ sec/m^3$$

$$X/Q_{\gamma} = 7.86\ (-5)\ sec/m^3$$

The isolation is postulated to occur at 17 seconds following the break, which is 7 seconds greater than the maximum P/V closure time allowed in existing PNPS technical specifications.

The above doses were calculated for the 20-inch valves. The recent installation of the qualified 8-inch valves precludes the overpressure of the SBGTS duct work and its subsequent failure. This means that the SBGTS remains capable of performing its designed function; therefore, doses associated with a pipe break in containment are consistent with the acceptable levels in both the Pilgrim Station Final Safety Analysis Report (FSAR) and 10 CFR 100.

BECo concurs that purging and venting should be minimized during reactor operation. Nevertheless, there are operation needs which dictate opening the valves, such as de-inerting prior to drywell entry. The 90 hour limit Pilgrim administratively imposed was an interim step taken because assurance of valve closure was not conclusively demonstrated for the Rockwell valves. Since the 8" valves have been installed, tested, and the design found acceptable by NRC, and since the valves are capable of closing against a DBLOCA, the need for such an interim step has been removed, and therefore the need for a 90 hour "window" is removed.

The minimization of P/V valve open time is inherent in the operational requirements of Pilgrim, for the following reasons:

(1) 10CFR50.44(C)(3)(i) requires that all BWR's must have an inerted atmosphere; such could not be maintained if the 8" valves were open for long periods of time;

(2) A pressure differential between the torus and the drywell, in accordance with PNPS technical specification 3.7.A.1.i, would be difficult-to-impossible to maintain with the valves open for extended periods.

(3) Section 5.2.3.7 of the FSAR discusses the mission of drywell purging, which is to "facilitate personnel access." The purge valves are also occasionally used to allow nitrogen make-up, or to control drywell pressure. Both tasks require relatively short open times;

(4) PNPS technical specification 3.7.A.5.a states: "After the completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5% oxygen by volume with nitrogen gas during reactor power operation." Technical specification 3.7.A.5.b gives a 24 hour time frame in which to attain 5% oxygen subsequent to placing the mode switch in run. Technical specification 3.7.A.5.b also allows the commencement of de-inerting 24 hours prior to shutdown. While these technical specifications do not specifically set a time limit for valve open time, they do indirectly accomplish this because to maintain the containment inert the valves could not be open for very long; thus the technical specifications serve to minimize valve open time consistent with the purge system's mission as described in the FSAR, and with the dictates of plant operation.

As shown above, the goal of minimizing P/V valve open time during power operation is already adequately addressed, removing the need for a discrete numerical value, such as 90 hours. It is our intention to remove the 90 hour restriction and achieve minimization through conformance with the dictates of plant design requirements, code requirements (10CFR50.44(C)(3)(i)), and existing technical specifications.

Item 3

"Regarding the four two-inch isolation valves that vent to the Standby Gas Treatment System, the licensee should either: (1) show that the low-low water level isolation signal provides all the protection required when the normal isolation signals are bypassed, (2) remove the bypass, or (3) modify the design to satisfy Criterion 1."

Response

The input signals to the 2" normal drywell and torus vent and purge valves that cause them to automatically shut are high drywell pressure, low reactor water level, low-low reactor water level and refueling floor exhaust high radiation. Of these input signals only high drywell pressure and low-low reactor water level are safety actuation signals. BECo does not consider low reactor water level and refueling floor exhaust high radiation safety actuation signals for the reasons described in the following narrative:

Low Reactor Water Level

PNPS plant design change 80-03 installed the low-low reactor water level isolation signal to the 2" valves. The low-low reactor water level setpoint is 6.44 ft. above the top of the active fuel. The addition of the low-low reactor water level signal removed the need for the low reactor water level signal. No core damage will result if reactor water

level remains above the low-low reactor water level setpoint during a transient; therefore containment isolation is not required above this level. The low-low reactor water level isolation signal cannot be overridden and is a lock-in signal. All the 2" normal purge and vent valves must be reset to the shut position, even if the low-low reactor water level clears, before any valve can be opened. The low reactor water level signal is maintained as a backup for the low-low reactor water level, but is not required and could be removed without reducing safety.

Refueling Floor Exhaust High Radiation

In a safety evaluation attached to a letter of October 14, 1981 from Mr. D. G. Eisenhut of the NRC to Mr. J. T. Dente of the BWROG concerning T.A.P. II.E.2.4(7) the following was stated:

> For containment vent and purge lines that are three inches or less in diameter and used for periodically relieving the pressure buildup inside the containment, no high radiation signal is required. This position is justified by the lower radiological consequences for these small lines and because the valves in these lines can be expected to close very reliably and quickly under accident condition.

Since the subject lines are less than 3" the refueling floor exhaust high radiation signal need not be considered a safety actuation signal for this application.

The control switches for the 2" normal drywell and torus purge and vent valves are of three position design: open, shut, and emergency open. A key is required to position a switch to the emergency open position. When a control switch is taken to the emergency open position, the high drywell pressure, low reactor water level and refueling floor exhaust high radiation isolation signals are overridden. The reasons given above indicate that only one safety actuation signal, high drywell pressure, is overridden. The low-low reactor water level isolation signal cannot be overridden and the other two signals are not safety actuation signals.

Because only one safety actuation signal is ever overridden, the NRC's criteria are met. Further, the ability to vent the containment after a non-LOCA containment high pressure trip is desirable to prevent or mitigate the consequences of a plant transient. For these reasons we believe the existing system logic is acceptable and plan no further action concerning it.

NUREG 0737, Item II.E.4.2(7)

"Your letter dated October 25, 1983 provided additional information concerning containment isolation on high radiation (Item II.E.4.2(7)). However, as the result of a telephone discussion with members of your staff on February 27, 1984, we are expecting further communication from you prior to our acting on your letter."

Response

Based on our discussion of February 27, 1984 BECo re-examined the NRC's position as well as our own on II.E.4.2(7). We also re-examined possible modifications which might provide a reliable method to isolate purge and vent valves on a high radiation signal.

We examined potential modifications and estimate that to appropriately satisfy the intent of II.E.4.2(7) would cost between $500,000 to $1,000,000.

As we have stated in the past, the existing isolation signals provide adequate protection to the public health and safety. Our October 25, 1983 submittal demonstrated that for PNPS no sufficiently reliable and appropriate monitor existed which could provide rapid isolation, and also reliably and consistently detect release rates of radioactive material such that the comparatively low offsite dose rates of 10 CFR 20 would be met.

This is because 10 CFR 20 restricts annual offsite doses to 500 mrem/yr to the whole body and 1500 mrem/yr to the thyroid (of a child or infant), and although a monitor might detect release rates that result in offsite dose rates equivalent to 500 mrem/yr whole body, it would not be capable of detecting release rates of halogens that result in offsite dose rates equivalent to 1500 mrem/yr to the thyroid. The maximum offsite dose rates which would result from a leakage of 25 gpm of spiked primary coolant into the drywell during purging operation, assuming 10% iodine partitioning and 50% plateout, are calculated to be about 0.06 mrem/hr to the whole body and about 2.0 mrem/hr to the thyroid. The corresponding maximum gamma dose rate on the surface of a 20" purge line, assumed to be 20 ft. in length, carrying the drywell atmosphere radionuclide concentration specified above, would be approximately 2.0 mrem/hr.

Therefore, it is our contention that a leakage rate greater than 125 gpm of primary coolant into the drywell could be experienced before such a monitor would indicate a reliably measurable increase above ambient background radiation levels, and that based on this, the drywell sump level, drywell humidity and temperature indication, and the existing drywell airborne radiation monitor are adequate to detect quantities of leakage which could result in significant offsite doses. We have provided this information in our letter of October 25, 1983, but have received no documented review of it from NRC other than a verbal reference to the original SER rejecting the Boiling Water Reactor Owners' Group position, a position which we believe did not present the objection to II.E.4.2(7) we presented in October 25, 1983 and herein again present.

In sum, we believe that the reduction in risk purported in II.E.4.2(7) has not been clearly demonstrated, and in light of the substantial costs involved in implementing an appropriate modification, we are convinced our resources could be more effectively employed in areas where significant, concrete risk reduction can be achieved.

Very truly yours,

W J Harrington

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