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July 23, 1984

United States Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. George W. Knighton, Chief
Licensing Branch 3
Office of Nuclear Reactor Regulation

SUBJECT: Beaver Valley Power Station - Unit No. 2
Docket No. 50-412
Response to Draft SER Open Item 102

Gentlemen:

The response to the NRC Accident Evaluation Branch's Draft SER Open Item No. 102 is provided in Attachment 1. The associated revisions to FSAR Section 15.6.2 and Tables 9.3-8, 15.0-12, 15.6-2, and 15.6-3 are provided in Attachment 2.

DUQUESNE LIGHT COMPANY

By E. J. Woolever
E. J. Woolever
Vice President

JDO/wjs
Attachments

cc: Mr. G. Walton, NRC Resident Inspector (w/a)
Mr. E. A. Licitra, Project Manager (w/a)
Ms. M. Ley, Assistant Project Manager (w/a)

SUBSCRIBED AND SWORN TO BEFORE ME THIS 23rd DAY OF July, 1984.

Anita Elaine Reiter
Notary Public

ANITA ELAINE REITER, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

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1/1

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

On this 23rd day of July, 1984, before me, a Notary Public in and for said Commonwealth and County, personally appeared E. J. Woolever, who being duly sworn, deposed and said that (1) he is Vice President of Duquesne Light, (2) he is duly authorized to execute and file the foregoing Submittal on behalf of said Company, and (3) the statements set forth in the Submittal are true and correct to the best of his knowledge.

Anita Elaine Reiter
Notary Public

ANITA ELAINE REITER, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

ATTACHMENT 1

Draft SER Open Item No. 102 (Section 15.6.2): Radiological Consequences of a Small Line Break DBA

Response:

The radiological consequences resulting from a small line break have been analyzed based on 40% of the reactor coolant flashing to steam upon entry into the building atmosphere. The assumption is made that the fraction of dissolved iodine fission products becoming airborne as gas and particulates is equal to the fraction of coolant flashing to steam.

All potential locations for the small line break in the auxiliary building are within ventilation zones of the supplementary leak collection and release system (SLCRS). A small line break in the contiguous areas would be serviced by the SLCRS after receipt of a high radiation signal from a QA Category II radiation monitor. However, the analysis does not take credit for SLCRS operation.

The results of the analysis are presented in the revisions to FSAR Section 15.6.2 and Tables 9.3-8, 15.0-12, 15.6-2, and 15.6-3 (Attachment 2). These revisions will be incorporated into a future FSAR amendment.

TABLE 9.3-8 (Cont)

Tube side

Design pressure (psig)	2,735
Design temperature (°F)	650
Material	Austenitic stainless steel

Operating parameters

Shell side

	Normal	Maximum Purification
Flow (lb/hr)	29,826	59,700
Inlet temperature (°F)	543.5 → 542.5	549
Outlet temperature (°F)	283 → 489.3	446 ← 287

Tube side

Flow (lb/hr)	22,370	52,250
Inlet temperature (°F)	130	130
Outlet temperature (°F)	489.3	446

<u>Letdown Orifices</u>

General

Design pressure (psig)	2,485
Design temperature (°F)	650
Normal operating inlet pressure (psig)	2195
Normal operating temperature (°F)	290
Material of construction	Austenitic stainless steel

60 gpm Orifice

Quantity	2
Design flow (lb/hr)	29,826
Differential pressure at design flow (psig)	1,900
Diameter (inches)	0.242

45 gpm Orifice

Quantity	1
Design flow (lb/hr)	22,370
Differential pressure at design flow (psi)	1,900
Diameter (inches)	0.215

TABLE 15.0-12

POTENTIAL DOSES DUE TO POSTULATED ACCIDENTS
(Rem)

Postulated Accident	FSAR Section	Exclusion Area Boundary Thyroid	Exclusion Area Boundary Whole Body Gamma	Exclusion Area Boundary Beta Skin	Low Population Zone* Thyroid	Low Population Zone* Whole Body Gamma	Low Population Zone* Beta Skin
Main steam line break	15.1.5						
Pre-accident iodine spike		5.5	6.2×10^{-2}	2.5×10^{-3}	3.1×10^{-1}	3.6×10^{-4}	1.7×10^{-4}
Failed fuel		2.3×10^{1}	2.2×10^{-1}	9.3×10^{-2}	1.4	1.8×10^{-3}	8.3×10^{-3}
Loss of nonemergency ac power to the station auxiliaries	15.2.6	1.5×10^{-1}	5.1×10^{-4}	4.0×10^{-4}	2.1×10^{-2}	6.5×10^{-5}	6.8×10^{-5}
Locked rotor	15.3.3	2.5×10^{-1}	9.5×10^{-4}	5.1×10^{-4}	1.7×10^{-1}	4.2×10^{-4}	1.7×10^{-4}
Rod ejection	15.4.8						
Containment leakage		4.0×10^{1}	1.8×10^{-1}	6.3×10^{-2}	2.0	9.2×10^{-3}	3.2×10^{-3}
Secondary side		2.1×10^{-1}	5.0×10^{-1}	3.6×10^{-1}	1.1×10^{-2}	2.5×10^{-2}	1.8×10^{-2}
Small line break - loss-of-coolant	15.6.2	9.8×10^{-1} 1.6×10^{1}	2.9×10^{-2} 6.8×10^{-2}	1.7×10^{-2} 2.3×10^{-2}	5.0×10^{-2} 8.2×10^{-1}	2.5×10^{-3} 3.4×10^{-3}	1.1×10^{-3} 1.2×10^{-3}
Steam generator tube rupture	15.6.3						
Pre-accident iodine spike		6.2	8.8×10^{-2}	7.0×10^{-2}	6.9×10^{-1}	4.8×10^{-3}	3.7×10^{-3}
Concurrent iodine spike		6.7	1.1×10^{-1}	7.4×10^{-2}	1.2	7.4×10^{-3}	4.3×10^{-3}
Loss-of-coolant	15.6.5						
Containment leakage		2.7×10^{1}	4.7	2.1	1.3×10^{1}	2.4×10^{-1}	1.1×10^{-1}
ECCS leakge		6.3×10^{-1}	4.5×10^{-3}	1.2×10^{-3}	4.9×10^{-1}	4.9×10^{-3}	1.6×10^{-3}
Waste gas system rupture	15.7.1	-	1.7×10^{-1}	1.6×10^{-1}			
Fuel handling	15.7.4	2.4×10^{1}	3.6	6.1			

NOTE:

*For duration of accident

limit value throughout the transient; thus, the departure from nucleate boiling (DNB) design-basis as described in Section 4.4 is met.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

Lines connected to the RCS and penetrating the containment, as well as isolation provisions are identified in Table 6.2-60.

There are no instrument lines connected to the RCS that penetrate the containment. There are, however, the sample lines from the hot and cold legs of reactor coolant loops and the steam and liquid space of the pressurizer, and the CVCS letdown and excess letdown lines that penetrate the containment. The sample lines and the CVCS letdown and excess letdown lines are all provided with normally open containment isolation valves on both sides of the containment wall. In all cases, the containment isolation valves are designed in accordance with the containment isolation requirements of General Design Criterion 55 (Section 6.2.4).

The most severe pipe rupture with regard to radioactivity release during normal BVPS-2 operation is a complete severance of the 2-inch letdown line at a location outside containment, upstream of the letdown heat exchanger. This event would result in a loss-of-reactor coolant at the rate of approximately 160 gpm based on a density of 57 lbs/ft^3 and on the flow restriction provided by two of the three letdown line orifices in service (the 45-gpm orifices and one of the 60-gpm orifices), shown on Table 9.3-8 and Figure 9.3-24.

15

The time required for the operator to identify the accident and isolate the rupture is expected to be less than 30 minutes. Diverse instrumentation in the form of letdown line pressure downstream of the postulated break location, volume control tank level and pressurizer level with indication at the main control board will allow detection of the failure by the operator. The operator would isolate the letdown line rupture by closing the letdown orifice isolation valves 2CHS*AOV200A, B, and C followed by closing the pressurizer low level isolation valves 2CHS*LCV460A and B. The operator would also close the letdown line containment isolation valve 2CHS*AOV204, to isolate the rupture. All valves are provided with control switches with indicating lights at the main control board and at the emergency shutdown panel. All valves are air-operated and designed to fail closed on loss of air or electrical power. There are no single failures that would prevent isolation of the letdown line rupture.

and flow

Insert "A" (pg. 15.6-4a)

Insert "B" (pg. 15.6-4a)

INSERT "A"

In addition, a control room operator can determine specific plant areas which are experiencing high radiation after receiving plant high radiation annunciation.

INSERT "B"

one of the pressurizer low level isolation valves, 2CHS*LCV460B, and by closing the letdown line containment isolation valve, 2CHS*AOV204. Following isolation of the rupture, the operator would also close 2CHS*LCV460A and the letdown orifice isolation valves, 2CHS*AOV200A, B, and C.

15.6.2.2 Analysis of Effects and Consequences

Method of Analysis

The amount of primary coolant released is conservatively estimated by assuming critical flows in the ruptured letdown line. The mass of fluid released from the postulated break was calculated using the Zaloudek correlation in WCAP-8312A (Westinghouse 1975a) for subcooled liquids and the theoretical model developed by Moody for saturated conditions. Immediately after the rupture, the Moody model is used for a saturated liquid until the liquid in the letdown line between the orifices and rupture point is depleted. After the liquid is depleted, Zaloudek's subcooled correlation is used at the orifice and continues for 30 minutes until isolation occurs. These critical flow correlations are in accordance with WCAP-8312A (Westinghouse 1975a). The assumptions used for the analysis are summarized in Table 15.6-2. The auxiliary building pressure transient for the postulated break is shown on Figure 3.11-3Q.

at 15 minutes after the break

15.6.2.3 Radiological Consequences

The failure outside the containment of small lines carrying primary coolant is postulated to occur in the letdown line to the letdown heat exchanger. in the auxiliary building. The rupture of this line will result in the loss of primary coolant, with isolation occurring within 30 15 minutes. The rupture will result in the discharge of primary coolant directly into the auxiliary building, or into the contiguous areas with the radioactivity released to the environment at ground level. Insert "C" (pg. 15.6-5a)

The assumptions for evaluating the radiological consequences of the postulated small line failure are summarized in Table 15.6-2. The conservative analysis assumes primary coolant Technical Specification equilibrium activities as presented in Table 15.0-8.

Additionally, a concurrent iodine spike is postulated to occur with iodine release rates into the primary coolant as shown in Table 15.0-10. The resulting releases to the environment based on the stated assumptions and postulated activities are presented in Table 15.6-3.

The radiological consequences resulting from a postulated failure of a small line carrying primary coolant outside containment are presented in Table 15.0-12. The offsite doses are determined using the calculated environmental releases for this accident and the atmospheric dispersion values given in Table 15.0-11. The methodology for calculating the offsite doses is discussed in Appendix 15A. The radiological consequences for this event are a small fraction of the guidelines of 10 CFR 100 that is less than 2.5 Rem whole body and 30 Rem thyroid.

INSERT "C"

All potential locations for the small line break in the auxiliary building are within ventilation zones of the supplementary leak collection and release system (SLCRS). A small line break in the contiguous areas would be serviced by the SLCRS after receipt of a high radiation signal from a QA Category II radiation monitor. However, the conservative analysis does not take credit for SLCRS operation.

TABLE 15.6-2

PARAMETERS USED FOR THE SMALL LINE CARRYING PRIMARY COOLANT FAILURE

Characteristics	Expected	Technical Specification
Power (MWt)	2,766	2,766
Fraction of failed fuel	0.0012	0.0026
Line failure	Letdown line to inlet of letdown heat exchanger	
Break size (in)	2	2
Time required to detect and isolate failure (hr) min	0.5 ← 15	0.5 ← 15
Total amount of primary coolant released (lb)	36,500 ← 20	36,500 ← 20
Temperature of released primary coolant (°F)	287	287 ← 549
Fraction of iodine assumed airborne from pipebreak	0.1	0.1 ← 0.4
Supplementary leak collection and release system iodine filter efficiency (%)	95	95 ← 0
Primary coolant concentrations	Table 11.1-2	Table 15.0-8
Iodine spiking - release rates (assumed to occur for duration of accident)	Table 15.0-10	Table 15.0-10

Primary coolant release rate from break (lbm/sec)

TABLE 15.6-3

SMALL LINE FAILURE RELEASES TO THE ENVIRONMENT

Nuclide	Releases (Ci) 0-2 Hr		2 Hr-30 Days*
Kr-83m	1.9	9.2×10^{-1}	
Kr-85m	9.1	4.5	
Kr-85	4.8×10^{1}	2.4×10^{1}	
Kr-87	5.2	2.6	
Kr-88	1.4×10^{1}	6.9	
Kr-89	4.4×10^{-1}	2.2×10^{-1}	
Xe-131m	4.7×10^{-1}	2.3×10^{-1}	8.3×10^{-2}
Xe-133m	1.3×10^{1}	6.6	2.9×10^{-1}
Xe-133	1.1×10^{2}	5.6×10^{1}	4.0
Xe-135m	4.0×10^{1}	3.6	1.6×10^{2}
Xe-135	1.6×10^{1}	7.0	1.0×10^{1}
Xe-137	7.1×10^{-1}	3.5×10^{-1}	
Xe-138	2.9	1.5	
I-131	7.1×10^{-1}	1.2×10^{1}	
I-132	3.4	2.0×10^{1}	
I-133	1.5	2.7×10^{1}	
I-134	1.4	2.7×10^{1}	
I-135	1.2	2.3×10^{1}	

NOTE:

*Due to decay of iodines collected on SLCRS filters.