Duquesne Light

Nuclear Construction Division
Robinson Plaza, Building 2, Suite 210
Pittsburgh, PA 15205

2NRC-4-122
(412) 787-5141
(412) 923-1960
Telecopy (412) 787-2629
August 10, 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
Reactor Systems Branch Open Items

Gentlemen:

This letter forwards responses to draft SER open items provided by the Reactor Systems Branch (RSB). This draft SER material, which was officially transmitted from the NRC to Duquesne Light Company on May 30, 1984, and June, 8, 1984, contains the following open items: 103, 108 through 111, 158 through 163, and 200 through 203.

Informal response to all of these open items were transmitted to you on July 20, 1984.

All fifteen of the RSB draft SER open items have been addressed.

DUQUESNE LIGHT COMPANY

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

JJS/wjs
Attachments

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

SUBSCRIBED AND SWORN TO BEFORE ME THIS
10th DAY OF August, 1984.

Elva G. Lesondak
Notary Public

ELVA G. LESONDAK, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

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COMMONWEALTH OF PENNSYLVANIA )
) SS:
COUNTY OF ALLEGHENY )

On this 10th day of August, 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 Companv, and (3) the statements set forth in the Submittal are true and correct to the best of his knowledge.

Elva G. Lesondak
Notary Public

ELVA G. LESONDAK, NOTARY PUBLIC
ROBINSON TOWNSHIP, ALLEGHENY COUNTY
MY COMMISSION EXPIRES OCTOBER 20, 1986

OPEN ITEM 103:

In response to the staff's concern that 30 minutes is not sufficient to diagnose and isolate a steam generator tube rupture, the applicant has provided additional data regarding the system's response and radiological consequences after a steam generator tube rupture accident. This information, however, did not support the isolation time of the affected steam generator ac 30 minutes.

Upon receipt of additional information, the staff will complete the review of the consequences of this accident and provide our evaluation.

RESPONSE:

Refer to the response submitted for this open item in letter 2NRC-4-086 from E. J. Woolever to G. W. Knighton dated June 20, 1984.

OPEN ITEM 108:

II.K.1.5 Review ESF Valve Positions, Controls, and Related Test and Maintenance Procedures to Assure Proper ESF Functioning

II.K.1.10 Review and Modify Procedures for Removing ESF from Service to Assure Operability Status is Known

The applicant states that the intent of these two items will be met when the Operating and Maintenance Procedures are written. They are scheduled to be completed in June, 1985. The acceptability of the measures taken to satisfy these items will be evaluated when these procedures are submitted.

II.K.3.17 Report on Outages of ECCS

The applicant states that the intent of these two items will be met when the Operating and Maintenance Procedures are written. They are scheduled to be completed in June, 1985. The acceptability of the measures taken to satisfy these items will be evaluated when these procedures are submitted.

RESPONSE:

As stated above, the Operating and Maintenance Procedures will meet the intent of these NUREG 0737 action items. This item should be considered confirmatory.

OPEN ITEM 109:

II.K.2.13 Thermal Mechanical Report: Effect of High-Pressure Injection on Vessel Integrity for Small-Break LOCA with no Auxiliary Feedwater

Staff review of this item will be covered in NRC unresolved safety issue A-49, "Pressurized Thermal Shock."

RESPONSE:

NRC review is required. No further DLC action is required.

OPEN ITEM 110:

II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

As a response to Item II.K.3.2, the applicant referenced a generic Westinghouse Owners Group submittal. Should staff generic review of this matrial conclude otherwise, NRC will request further consideration of modification of Beaver Valley Unit 2.

RESPONSE:

NRC review is required. No further DLC action is required.

OPEN ITEM 111:

II.K.3.5 Automatic Trip of RCP's During LOCA

In response to this criterion, the applicant stated that Westinghouse performed an analysis of delayed RCP trip during LOCA. This analysis is documented and is the basis for the Westinghouse position on RCP trip (i.e., automatic RCP trip is not necessary because sufficient time is available for manual tripping of the RCP's).

Westinghouse has submitted a generic report which is under review. The applicant should state whether or not it intends to endorse this report and comply with the criteria proposed in it assuming the NRC finds it acceptable.

II.K.3.30 Revised Small-Break LOCA Methods to Show Compliance with 10CFR50; Appendix K

In response to this criterion, the applicant stated that Westinghouse has submitted a new small-break evaluation model to NRC. The staff is currently reviewing this submittal.

II.K.3.31 Plant-Specific Calculations to Show Compliance with 10CFR50.46

The applicant states that the present (i.e., July, 1983) Westinghouse small-break, loss-of-coolant evaluation model was used for the analyses which are discussed in FSAR Section 15.6.5. However, this does not constitute a review that shows Beaver Valley Unit 2 is in full compliance with 10CFR50.46. After the staff's review of this evaluation model is completed, a specific submittal of this issue will be required.

RESPONSE:

NRC review is required.

For II.K.3.5, refer to Letter 2NRC-4-005 from E. J. Woolever to D. G. Eisenhut dated January 19, 1984, which addresses Generic Letter 83-10C and endorses the Westinghouse Owners Group generic report. It also states that the necessary information will be incorporated in the BVPS-2 Emergency Operating Procedures.

For II.K.3.30 and II.K.3.31, DLC will be submitting a letter in accordance with Generic Letter 83-35 which will state that the Westinghouse Owners Group generic model studies demonstrating the necessary conservatism are applicable to BVPS-2.

OPEN ITEM 158:

Low-temperature overpressure protection is primarily provided by two of the three pressurizer PORVs. These two have their opening setpoints automatically adjusted as a function of reactor coolant temperature. The reactor coolant temperature measurements will be auctioneered to obtain the lowest value. This temperature will be translated into a PORV setpoint curve that will adequately account for the lag in the temperature change of the reactor vessel and for possible single failures in the auctioneering system, so the system pressure will always be below the maximum allowable pressure. This PORV setpoint curve shall be periodically updated, as shall be specified in the bases for the technical specifications, to ensure that the stress intensity factors for the reactor vessel at any time in life are lower than the reference stress intensity factors as specified in 10 CFR 50, Appendix G.

The applicant will provide PORV setpoint values later, and the staff will report its evaluation of these in a supplement to this SER.

Subject to the generation of a conservative PORV setpoint curve and appropriate Technical Specifications, the staff concludes that the overpressure protection system meets the relevant criteria of GDC 15 and is, therefore, acceptable. Conformance to Appendix G to 10 CFR 50 criteria will be confirmed when the PORV setpoint curve is found acceptable.

RESPONSE:

Westinghouse will provide Duquesne Light with PORV setpoints in March, 1986. These will be in the form applicable for use in BVPS-2 Technical Specifications.

OPEN ITEM 159:

GDC 19 states that a control room shall be provided from which actions can be taken to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents. SRP 5.4.7 stipulates that the control of the RHRS be such that the cooldown function can be performed from the control room assuming a single failure of any active component, with only either onsite or offsite electric power available. Any operation required outside of the control room is to be justified by the applicant.

RESPONSE:

The NRC has deleted this open item based on its finding that the actions outside of the control room that could be taken, as described in the FSAR, are acceptable.

OPEN ITEM 160:

For RHR's with automatic isolation, Branch Technical Position RSB 5-1 criteria calls for adequate pressure relief capacity while the isolation valves are closing. The applicant states in FSAR Amendment 3 that additional pressure relief capacity is provided by the low-temperature overpressure protection system and that an evaluation to determine the adequacy of the RHRS overpressure protection system will be available by March 31, 1984. We will determine the adequacy of the RHRS pressure relief when this evaluation is received.

RESPONSE:

See revised response to Q440.18 Amendment 6.

OPEN ITEM 161:

Since both RHRS pumps are located inside of containment there is a question of whether or not this environment could cause a common mode failure. Moreover, the Equipment Environmental Qualification Table (3.11-1) in the FSAR for the RHRS does not include the RHR pumps. For a reliable system these pumps are going to have to be qualified for the containment environment and included in Table 3.11-1.

RESPONSE:

Use of the RHR system is considered to be a reliable and the preferred means of residual heat removal but not the method used to meet GDC 34. A further description of this method is provided in Appendix 5A.

RHR system reliability considerations are discussed in FSAR Section 5.4.7.2.6. The RHR pump motors which are considered as Safety Class 2 in FSAR Table 3.2-1, are not required to safely shut down after a design basis accident or to prevent or mitigate the consequences of such an accident. Therefore, they are not environmentally qualified for design base accident conditions inside containment. In the event the RHR system is unavailable, the safety grade systems discussed in FSAR Section 5.4.7.4.6 are qualified to perform the function of removing residual heat from the reactor core for normal, abnormal, and design base accident conditions. These safety-grade systems include the ECCS, AFWS along with the steam generator (S/G) safety valves, and S/G PORV's. Class IE safety-related equipment associated with the ECCS, AFWS, and S/G PORV's are identified in Table 3.11-1. The use of these safety-grade systems conform with BTP RSB 5-1 as applicable to BVPS-2. While the required safe shutdown design basis for BVPS-2 is hot standby, a considerable number of modifications to plant design have been implemented since issuance of the construction permit to enhance BVPS-2 cold shutdown capability. This cold shutdown capability has been evaluated and is discussed in Appendix 5A. Table 3.11-1 includes all Class IE electrical equipment necessary to remove residual heat from the reactor.

Based on the above, BVPS-2 meets GDC-34 by providing systems to transfer fission product decay and other residual heat from the reactor core at a rate within acceptable design limits.

OPEN ITEM 162:

The adequacy of the mixing of borated water added to the RCS under natural circulation and the ability to cooldown Beaver Valley Unit 2 with natural circulation will be verified by referencing the results of a natural circulation test at a similar plant. For this type of verification a detailed comparison of the two plants is required. This must include a comparison of the elevations of the major components.

RESPONSE:

Beaver Valley and North Anna have been compared to ascertain any differences that could potentially affect natural circulation flow and attendant boron mixing. Because of the similarity between the plants, it was concluded that the natural circulation capabilities would be similar, and therefore, the results of prototypical natural circulation cooldown tests conducted at North Anna are representative of the capability of Beaver Valley Unit No. 2.

OPEN ITEM 163

In response to the staff's questions on an inspection program, operator training, and emergency procedures for dealing with debris, vortices, air entrainment, and other containment sump problems, the applicant stated in FSAR Amendment 3 that a response would be provided in a later amendment. This item will be considered open until that time.

RESPONSE:

The question referred to in this open item is 440.35 which was answered in Amendment 5, February 1984.

OPEN ITEM 200:

For each event analyzed, the worst operating conditions and the most limiting single failure were assumed, and credit was taken for minimum engineered safeguards response. In questions 440.73 and 440.74 the staff has asked the applicant to:

1. Supply listings of the single failures which were assumed for each event in the Chapter 15 analyses.

2. Supply the limiting single failure that results in the peak pressure or limiting performance for each event.

3. Show the effect of a loss of offsite power on all anticipated operational occurrences and postulated accidents.

When this information is received it will be incorporated into the evaluations of the individual events.

RESPONSE:

Refer to the response to Question 440.1, Amendment 2, July 1983 and the response to Question 730.1, Amendment 3, October 1983, Item A-44. Responses to Questions 440.73 and 440.74, to be included in Amendment 8, will refer to these earlier questions and responses.

OPEN ITEM 201:

Assuming offsite power is available to run the reactor coolant pumps, the applicant analyzed the turbine trip event for a complete loss of steam load from full power without a direct reactor trip and with only the pressurizer and steam generator safety valves assumed for pressure relief. These assumptions result in the highest peak RCS pressure for any "decreased heat removal" event. The calculated peak value is 2560 psia, which is well below the ASME limit of 110% of the design pressure. For these assumptions the minimum DNBR is 1.75, which is well above the minimum limiting value of 1.30.

The applicant's analyses show that if instead of relying on just the safety valves, the pressurizer spray and PORV's are used to limit the pressure during this turbine trip event, the minimum DNBR can go down to 1.60. If a stuck open PORV were to be assumed as the single failure during this course of action, it appears that the DNBR could go lower. The applicant has not discussed the possibility of a stuck open PORV or atmospheric steam dump valve being the worst single failure during this course of action.

RESPONSE:

Refer to response to Question 440.1, Amendment 2, July 1983, under Turbine Trip (Section 15.2.3).

OPEN ITEM 202:

In response to a question on a loss of offsite power (LOOP) during these events (RCP rotor seizure), the applicant states that a LOOP will have only a negligible effect on the critical parameters of RCS pressure and clad temperature and that it would have no effect whatsoever on the conclusions. The staff finds that a quantitative analysis of the worst case, which would have only two loops in operation, with a concurrent loss of offsite power is needed for the evaluation of this issue.

RESPONSE:

The locked rotor t for BVPS-2 has been reanalyzed with a loss of offsite power assumed. The results are affected as follows: Peak Reactor Coolant System pressure increases from 2608 psia to 2652 psia (2647 to 2679 for N-1), peak clad temperature increases from 1897 F to 1956 F (1773 to 1816 for N-1) and peak zirconium reaction increases from 0.39% to 0.57% (0.33 to 0.43 for N-1). Since all these are within the previously identified safety limits, the conclusions remain valid.

OPEN ITEM 203:

In response to a question on protection from inadvertent boron dilution during refueling, the applicant stated that during refueling the RCS is isolated from the potential source of unborated water. This isolation is accomplished by having the operators place danger tags on the primary grade water header isolation valves, or by locking these valves closed whenever the RCS water is below the normal level. The operator performing these tasks is required to sign off on each step of a procedural checklist. This long term use of administrative controls to prevent an inadvertent boron dilution during refueling has not been accepted by the staff on other plants and will be evaluated. The staff is not, at this point, convinced that a design basis event can be eliminated from detailed evaluation based on administrative means alone. We will report the resolution of this issue in a subsequent safety evaluation.

RESPONSE:

As stated in the response to Question 440.59, Amendment 3, it is DLC's intention to employ administrative controls to prevent the primary grade water header from providing unborated water addition to the RCS makeup system. Plant operators will be required to follow procedures which will instruct them to place danger tags on the primary grade water header isolation valves wherever the reactor coolant system is drained down below normal water level conditions.