Docket File
50-334

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 30, 1996

Mr. J. E. Cross
President-Generation Group
Duquesne Light Company
Post Office Box 4
Shippingport, PA 15077

SUBJECT: STAFF EVALUATION REPORT OF BEAVER VALLEY POWER STATION, UNIT NO. 1, INDIVIDUAL PLANT EXAMINATION (IPE), INTERNAL EVENTS, GENERIC LETTER (GL) 88-20 (TAC NO. M74378)

Dear Mr. Cross:

The purpose of this letter is to transmit our Staff Evaluation Report (SER) of Duquesne Light Company's (DLC's) IPE submittal for internal events and internal flood for Beaver Valley Power Station, Unit No. 1 (BVPS-1). The BVPS-1 IPE was submitted on October 1, 1992, in response to GL 88-20, and supplemented on March 10, 1995, in response to our January 5, 1995, request for additional information. Also included with our SER are our contractors' Technical Evaluation Reports (TERS).

The NRC staff performed a "Step 1" review which examined the IPE results for their "reasonableness" considering the design and operation of BVPS-1. The NRC staff employed Science & Engineering Associates, Inc., Concord Associates, and Scientech Inc. to review the front-end analysis, human reliability analysis, and back-end analysis, respectively, of the IPE submittal. Their TERs are attached as Appendices A, B, and C, respectively, to the SER. These contractor TERs were reviewed by the IPE "Senior Review Board" (SRB) as part of the NRC Office of Nuclear Regulatory Research (RES) quality assurance process. The SRB is comprised of RES staff and consultants at Sandia and Brookhaven National Laboratories with probabilistic risk analysis expertise.

The IPE has estimated a core damage frequency of 2.1E-4/reactor-year, including a contribution from internal flooding of 3E-6/reactor-year. Reactor coolant pump (RCP) seal loss of coolant accident contributes 46%, station blackout (SBO) 30%, containment bypass/isolation failures 21%, anticipated transients without scram 20%, and loss of emergency switchgear heating ventilation and air conditioning (HVAC) 16%.

The DLC defined vulnerabilities as "the fundamental contributors to risk" in the important scenarios. Based on this definition, the IPE identified eight vulnerabilities: (1) ac power generation capability, (2) reactor trip breaker failure, (3) pressurizer power-operated relief valve (PORV) block valve alignment, (4) loss of emergency switchgear room HVAC, (5) RCP seal cooling for SBO, (6) battery capacity for steam generator level during SBO, (7) pressurizer PORV sticking after loss of offsite power, and (8) fast 4160 V bus transfer failures. Plant improvements, however, were identified and considered for implementation.

DF01 1/1

Based on the "Step 1" review, we conclude that DLC has met the intent of GL 88-20. We do not recommend that a "Step 2" review be conducted. It is important to note that the NRC staff's review is not intended to validate the accuracy of DLC's IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on DLC's ability to examine BVPS-1 for severe accident vulnerabilities, and not specifically in the detailed findings (or quantification estimates), which stemmed from the examination.

With this letter, the NRC staff is closing TAC No. M74738.

Sincerely,
/S/
Donald S. Brinkman, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure: Staff Evaluation Report

cc w/encl: See next page

DISTRIBUTION
Docket File
PUBLIC
PDI-2 Reading
SVarga
JZwolinski
JStolz
DBrinkman
MO'Brien
OGC
ACRS
EChow
RHernan
EButcher
WHodges
PEselgroth, RGN-I

OFFICE	PDI-2/LA	PDI-2/PM	PDI-2/D		
NAME	MO'Brien	DBrinkman:rb	JStolz		
DATE	9/27/96	9/27/96	9/27/96		

OFFICIAL RECORD COPY
DOCUMENT NAME: BV74378.GEN

Based on the "Step 1" review, we conclude that DLC has met the intent of GL 88-20. We do not recommend that a "Step 2" review be conducted. It is important to note that the NRC staff's review is not intended to validate the accuracy of DLC's IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on DLC's ability to examine BVPS-1 for severe accident vulnerabilities, and not specifically in the detailed findings (or quantification estimates), which stemmed from the examination.

With this letter, the NRC staff is closing TAC No. M74738.

Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure: Staff Evaluation Report

cc w/encl: See next page

J. E. Cross
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

cc:

Jay E. Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW.
Washington, DC 20037

R. K. Brosi, Manager
Nuclear Safety Department (BV-A)
Duquesne Light Company
Beaver Valley Power Station
PO Box 4
Shippingport, PA 15077

Commissioner Roy M. Smith
West Virginia Department of Labor
Building 3, Room 319
Capitol Complex
Charleston, WVA 25305

John D. Borrows
Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
Post Office Box 3321
Harrisburg, PA 17105-3321

Ohio EPA-DERR
ATTN: Zack A. Clayton
Post Office Box 1049
Columbus, OH 43266-0149

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
ATTN: R. Barkanic
Post Office Box 2063
Harrisburg, PA 17120

Mayor of the Borrough of
Shippingport
Post Office Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 298
Shippingport, PA 15077

Duquesne Light Company
Beaver Valley Power Station
PO Box 4
Shippingport, PA 15077
ATTN: S. C. Jain, Vice President
Nuclear Services (BV-A)

STAFF EVALUATION REPORT

OF

BEAVER VALLEY POWER STATION, UNIT NO. 1

INDIVIDUAL PLANT EXAMINATION (IPE)

(INTERNAL EVENTS ONLY)

ENCLOSURE

I. INTRODUCTION

On October 1, 1992, Duquesne Light Company (DLC) submitte. he Beaver Valley Unit 1 (BV1) Individual Plant Examination (IPE) submittal in response to Generic Letter (GL) 88-20 and associated supplements. On January 5, 1995, the staff sent questions to the licensee for more information. The licensee responded in a letter dated March 10, 1995.

A "Step 1" review of the BV1 IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Scientech, Inc., and Concord Associates in the front-end, back-end, and human reliability analysis (HRA), respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the BV1 design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of contractors' findings is provided below. Details of the contractors' findings are in the attached technical evaluation reports (Appendices A, B, and C) of this staff evaluation report (SER).

In accordance with GL 88-20, BV1 proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." No other specific USIs or generic safety issues were proposed for resolution as part of the BV1 IPE.

The BV1 IPE submittal does not state that a "living" PRA will be maintained, however, the BV1 IPE submittal states: "DLC now recognizes the benefits of a PRA and the capability that has been developed will be maintained. This capability will support a comprehensive risk management program."

II. EVALUATION

BV1 is a Westinghouse 3-loop PWR with a large dry subatmospheric containment. The BV1 IPE has estimated a core damage frequency (CDF) of 2.1E-4/reactor-year from internally initiated events, including the contribution from internal floods. The BV1 CDF compares reasonably with that of other Westinghouse 3-loop PWR plants. Reactor coolant pump (RCP) seal loss of coolant accident (LOCA) contributes 46%, station blackout (SBO) 30%, containment bypass/isolation failures 21%, anticipated transients without scram (ATWS) 20%, and loss of emergency switchgear heating ventilation and air conditioning (HVAC) 16%. The licensee's Level 1 analysis appeared to have examined significant initiating events and dominant accident sequences.

The licensee made two changes since completion of the IPE, namely, installation of the station cross-tie and a reanalysis of ATWS sequences. These changes results in a new CDF of 1.2E-4/year.
Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of BV1 plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI

A-45, Decay Heat Removal Reliability, resolution.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operator actions as important in the estimate of the CDF: failure to set up portable fans to cool emergency switchgear, failure to restore electric power given SBO with auxiliary feedwater (AFW), failure to open normal switchgear ventilation supply louvers, failure to manually insert control rods, premature securing of safety injection, failure to align outside recirculation spray (RS) pump to low head safety injection (LHSI) for high pressure recirculation, failure to depressurize and cool down secondary in small LOCA (SLOCA). The staff concluded that there were limitations in the licensee's HRA approach. For example, human errors related to calibration of equipment were not appropriately treated in the HRA. Although it is unlikely that the omission of calibration errors critically impacts the licensee's overall conclusions from the IPE, the licensee may have missed the opportunity to identify potential enhancements to plant safety.

The licensee evaluated and quantified the results of the severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The licensee's back-end analysis appeared to have considered important severe accident phenomena. Among the BV1 conditional containment failure probabilities, early containment failure is 6.5% with high pressure melt ejection being the primary contributor, late containment failure is 43.4% with occurrence of containment overpressurization when containment heat removal is unavailable, bypass is 4.5% with steam generator tube rupture (SGTR) being the primary contributor, and containment isolation failure is 16.3% with a majority of contribution from emergency switchgear ventilation failures. The containment remains intact 29.3% of the time. Early radiological releases are dominated by loss of offsite power sequences and late releases are dominated by SBO sequences. The licensee's response to containment performance improvement program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified at BV1 by the licensee are:

1. Dedicated feedwater pump is powered from the emergency response facility diesel generator (DG) as an Appendix R backup for auxiliary feedwater.

2. There is an automatic switchover of emergency core cooling system (ECCS) from injection to recirculation.

3. The plant operates with two of the three PORV block valves closed. This feature tends to increase the CDF by reducing the pressure relief capability in response to an ATWS.

4. Ventilation to the emergency switchgear rooms is required. This feature tends to increase CDF by requiring the emergency switchgear ventilation to support 1E power operation.

The licensee defined a vulnerability as "the fundamental contributors to risk" in the important scenarios. Based on this definition, the IPE identified eight front-end vulnerabilities:

1. AC power generation capability,
2. Reactor trip breaker failure,
3. Pressurizer PORV block valve alignment,
4. Loss of emergency switchgear room HVAC,
5. RCP seal cooling for SBO,
6. Battery capacity for SG level during SBO,
7. Pressurizer PORV sticking after loss of offsite Power,
8. Fast 4160 V bus transfer failures.

In addition, the licensee identified containment overpressurization and containment bypass as two back-end vulnerabilities, based on large, early release frequencies.

Plant improvements were identified to address these vulnerabilities. These improvements, which were characterized by the licensee as shown below and have not been reviewed by the staff, are either implemented or under evaluation by the licensee:

1. Cross-tie DGs between Units 1 and 2,
2. Enhance procedures to de-power bus for enhanced recovery for ATWS,
3. Enhanced procedures to prevent overheating of emergency switchgear,
4. Use of high-temperature O-ring for RCP seal,
5. Enhanced procedures for load shedding and using portable battery chargers to extend time of SG level indication under loss of AC power,
6. Enhanced procedures and training to reduce 4160 V. breaker failure frequency,
7. Implement plant procedures and training to enhance the operator response to containment bypass sequences,
8. Use the diesel-driven fire system pump for some accident sequences.

For LOCA outside the containment, the licensee identified the importance of improving guidance to the operators on the key valve to close.

III. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regards to the information requested by GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the BV1 design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the BV1 IPE has met the intent of GL 88-20.

It should be noted, that the staff's review primarily focused on the licensee's ability to examine BV1 for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20. However, because the licensee intends to continue to use and maintain its IPE, the staff encourages the licensee to improve the BV1 IPE in order to make it a valuable tool for other regulatory applications.

Date: September 30, 1996

APPENDIX A

BEAVER VALLEY 1 NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

APPENDIX A

BEAVER VALLEY 1 NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)