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

May 13, 1993

Docket No. 50-412

Mr. J. D. Sieber, Senior Vice President
and Chief Nuclear Officer
Nuclear Power Division
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: STAFF EVALUATION OF BEAVER VALLEY UNIT 2 INDIVIDUAL PLANT EXAMINATION (IPE) - INTERNAL EVENTS, GENERIC LETTER 88-20 (TAC NO. M74379)

The purpose of this letter is to transmit our evaluation of your Individual Plant Examination (IPE) which you submitted on March 17, 1992, in response to Generic Letter 88-20.

Duquesne Light Company (DLC) responded to Generic Letter 88-20 and its supplements regarding Unit 2 in letters dated October 30, 1989, September 18, 1991, March 17, August 17, September 11, and October 26, 1992.

Our review examined the Beaver Valley Unit 2 IPE submittal (internal events only) and associated documentation which included the IPE summary report for the Beaver Valley Unit 2 Probabilistic Risk Assessment (PRA) and your responses to staff's questions. No specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the Beaver Valley Unit 2 IPE. A separate evaluation report will be issued to document the review of the external event portion of the Beaver Valley Unit 2 IPE after it is submitted.

Using DLC's definition of vulnerability, you identified seven "vulnerabilities" including AC power generation failure on station blackout (SBO), reactor coolant pump (RCP) seal LOCA on SBO, loss of emergency switchgear heating, ventilation and air conditioning (HVAC), 4160 V bus fast transfer failures, power operated relief valve (PORV) failure to reclose after loss of offsite power, battery capacity on SBO, and reactor trip breaker failure. DLC found that the contributions of these vulnerabilities to the core damage frequency (CDF) vary from 25% to 4%. We understand you are reviewing plant and procedure changes in order to address these vulnerabilities.

The staff notes that various characteristics of Beaver Valley Unit 2 plant design and operation, and of operator actions, were found to be important in the analysis of core damage frequency. Noteworthy among these are the following:

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(1) For station blackout sequences, both thermal barrier cooling and reactor coolant pump (RCP) seal injection are lost. The loss of all seal cooling could lead to seal failure and a potential loss of coolant accident (LOCA). The addition of a cross-tie connecting the 4kV normal busses of Beaver Valley Unit 1 and Beaver Valley Unit 2 would provide an alternate AC source. This modification is to be implemented to provide an acceptable station blackout coping capability using the emergency diesel generators of the other unit as the alternate AC source. DLC has committed to installing the necessary hardware, revising existing procedures, and providing training to effect the cross-tie capability. Additional modifications to address RCP seal integrity for loss of all seal cooling are under review. These modifications would include the consideration of new seal materials and alternate seal cooling systems. The modifications, if any, will be implemented in accordance with the resolution to Generic Issue 23.

(2) The important containment bypass sequences at Beaver Valley Unit 2 are initiated by a steam generator tube rupture (SGTR). Hence, operator response may be important. Specifically, operator action to facilitate isolation of the ruptured steam generator (by cool down and depressurization of the reactor coolant system (RCS)) may play an important role. Although substantial time is available for this action, it is desirable to have the emergency procedures for steam generator tube rupture events instruct the operators more explicitly on how to perform the depressurization for the sequences in which all high head safety injection is also failed. Procedures are being updated accordingly. Also, for SGTR events, the potential exists for a safety relief valve on the ruptured steam generator to stick open. Procedures and training are being improved to ensure that such a stuck-open valve would be locally gagged closed, thereby isolating the ruptured steam generator.

(3) Emergency switchgear ventilation is provided by a normally operating two-train fan system. Complete losses of such systems have been known to occur at other plants. The rooms served by emergency switchgear ventilation contain a number of heat loads. These rooms are situated so that simply opening doors will not produce a chimney effect. Thermal-hydraulic analyses indicate that equipment design temperature limits may be exceeded in less than one-half hour if all ventilation is lost. Loss of all emergency switchgear ventilation may lead to complete loss of all emergency AC power. The ability of the operators to provide alternate room cooling promptly to the switchgear areas is important. Currently, alarm response procedures inform the operators to investigate the cause of trouble, but do not provide explicit guidance on how to establish sufficient alternate cooling in the event that both emergency switchgear ventilation fan trains fail. The alarm response procedures are being reviewed to see if they can be enhanced to cover these scenarios.

(4) Beaver Valley Unit 2 is designed to try and stay on line following a load rejection accident. For a 100 percent load rejection accident, however, a high degree of reliability is not expected for a successful runback to house loads. Consequently, for a loss of offsite power event, the net effect of this design feature is just to delay the time of reactor trip. The delay is expected to lead to the lifting of the pressurizer PORVs. This actually has occurred during a loss of load test. In the event that the PORVs fail to reclose, the time available for electric power recovery from a station blackout event is significantly reduced. The option of eliminating the challenge by defeating the 100 percent load rejection capability is being considered.

The staff performed a "Step 1" review which is intended to determine whether or not a licensee's IPE process is capable of identifying significant core damage vulnerabilities. Based on our "Step 1" review, we conclude that you have met the intent of Generic Letter 88-20. We do not recommend that a "Step 2" review be conducted. It is important to note that the staff's review is not intended to validate the accuracy of your IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on your ability to examine Beaver Valley Unit 2 for severe accident vulnerabilities, and not specifically on the detailed findings (or quantification estimates) which stemmed from the examination.

With this letter, the staff is closing TAC NO. M74379.

Sincerely,
/S/
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Staff Evaluation

cc w/enclosure:
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G E Edison

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Division of Reactor Projects - I/II
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ENCLOSURE

STAFF EVALUATION OF BEAVER VALLEY UNIT 2
INDIVIDUAL PLANT EXAMINATION
(IPE)

(INTERNAL EVENTS ONLY)

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EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the Beaver Valley Unit 2 (BV-2) Individual Plant Examination (IPE) submittal and associated information. The latter includes licensee responses to staff generated questions seeking clarification of the licensee's process. No specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the BV-2 IPE.

The licensee's IPE is based on a BV-2 Level 2 probabilistic risk assessment (PRA), and a back-end analysis consistent with the staff's guidance contained in Generic Letter 88-20, Appendix 1 (Examination of Containment System Performance). Duquesne Light Company (DLC) personnel maintained involvement in the development and application of PRA techniques to the BV-2 facility, with the objective of transfer of PRA technology to the DLC personnel. The staff notes that virtually all of the plant departments provided input to the IPE/PRA development.

The licensee defined vulnerabilities as "the fundamental contributors to risk" in the important scenarios. These fundamental contributors are determined by delineating the sequence characteristics and then determining their importance by their respective contribution to core damage frequency (CDF) and release category frequency. The licensee identified seven "vulnerabilities" including AC power generation failure on station blackout (SBO), reactor coolant pump (RCP) seal LOCA on SBO, loss of emergency switchgear heating, ventilation and air conditioning (HVAC), 4160 V bus fast transfer failures, PORV failure to reclose after loss of offsite power, battery capacity on SBO, and reactor trip breaker failure. The licensee is reviewing plant and procedure changes proposed to address these vulnerabilities.

Based on the review of the BV-2 IPE submittal and associated documentation, the staff concludes that the licensee met the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in Generic Letter 88-20 and associated NUREG-1335 submittal guidance document; (2) the front-end systems analysis, the back-end containment performance analysis, and the human reliability analysis are technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) the licensee employed viable means (document review and walkdowns) to verify that the IPE reflected the current plant design and operation; (4) the PRA which formed the basis of the IPE had an extensive peer review; (5) the licensee participated fully in the IPE process consistent with the intent of Generic Letter 88-20; (6) the licensee appropriately evaluated BV-2's decay heat removal (DHR) function for vulnerabilities, consistent with the intent of the USI A-45 resolution; and (7) the licensee responded appropriately to recommendations stemming from the containment performance improvement (CPI) program. In addition, the licensee recognizes the potential benefits of a PRA and plans to use and maintain it.

It should be noted, however, that the staff's review is a process review which, in general, is not intended to validate the accuracy of the licensee's IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on the licensee's ability to examine BV-2 for severe accident vulnerabilities, and not specifically on the detailed findings (or quantification estimates) which stemmed from the examination.

I. BACKGROUND

On November 23, 1988, the NRC issued Generic Letter 88-20 (Ref. 1) which requires licensees to conduct an Individual Plant Examination in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to (1) develop an overall appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur at its plant, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335 (Ref. 2), all IPEs are to be reviewed by NRC teams to determine the extent to which each licensee's IPE process met the intent of Generic Letter 88-20. The IPE review itself is a two step process; the first step, or "Step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "Step 2" review. The decision to go to a "Step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PRA experience. A unique design may also warrant a "Step 2" to better understand the implication of certain IPE findings and conclusions. As part of this process, the BV-2 IPE only required a "Step 1" review.

On March 17, 1992, Duquesne Light Company (DLC) submitted the BV-2 IPE (Ref. 3 and 4) in response to Generic Letter 88-20 and associated supplements (Ref. 1, 5, and 6). The IPE submittal, based on the BV-2 PRA (PLG-0730 (Ref. 7)), consists of a Level 1 PRA, and a Level 2 containment performance assessment consistent with Generic Letter 88-20 Appendix 1. The IPE submittal contains the results of an evaluation of internal events, including internal flooding. The licensee plans to provide a separate submittal on findings stemming from the IPE for external events (IPEEE). The staff will review the IPEEE separately, within the framework prescribed in Generic Letter 88-20 Supplement 4 (Ref. 8).

On July 15, 1992, the staff sent a set of questions (Ref. 9) to the licensee seeking additional information and clarification. The licensee responded to the staff's request in letters dated September 11, 1992, and October 26, 1992 (Ref. 10 and 11).

The following list summarizes the basic information reviewed during the staff's evaluation of the licensee's IPE review process:

1. BV-2 response to Generic Letter 88-20 (Ref. 3)

2. BV-2 Probabilistic Risk Assessment Individual Plant Examination Summary Report (Ref. 4)

3. BV-2 responses (Ref. 10 and 11) to NRC request for additional information (Ref. 9)

This report documents findings and conclusions which stemmed from the NRC review. Specific numerical results and other insights taken from the licensee's IPE submittal are listed in the appendix.

II. STAFF'S REVIEW

1. Licensee's IPE Process

The BV-2 IPE submittal describes the approach taken by the licensee to confirm that the IPE represents the as-built, as operated plant. In addition to detailed document reviews by members of the PRA team (consultants and licensee personnel, some of whom are responsible for 10 CFR 50.59 safety evaluations), walk-throughs were performed for familiarization with plant/system operations, equipment layout for origin and susceptibility to floods and containment walk-throughs for information to be used for the back-end analysis.
Based on review of the information submitted with the IPE, the staff concludes that the licensee's walkdowns and documentation reviews constituted a viable process for confirming that the IPE represents the as-built, as-operated plant.

The IPE submittal contains a summary description of the licensee's IPE process, the licensee's personnel participation in the process, and the subsequent in-house peer review of the final product. The staff reviewed the licensee's description of the IPE program organization, composition of the peer review teams, and peer findings and conclusions. The staff notes the considerable participation of the DLC personnel in virtually all aspects of the IPE through technology transfer, model development, reviews, data collection, and requantification of the models with plant-specific data. In addition to the IPE team, other DLC departments were involved to insure that the models accurately portrayed the plant. The submittal indicated that, "DLC recognizes the potential benefit of the PRA and plans to use and maintain it."

As part of the IPE process DLC established an independent review team which consisted of personnel from all appropriate organizations including engineering, operations, training, and an independent safety engineering group. This review was in addition to internal reviews performed by DLC's consultants.

Based on the review of the IPE submittal and associated documentation, the staff concludes the licensee's peer review process provided reasonable assurance that the IPE analytic techniques had been correctly applied, and documentation was accurate.

The licensee defined vulnerabilities as "the fundamental contributors to risk" in the important scenarios. These fundamental contributors are determined by delineating the sequence characteristics and then determining their importance by their respective contribution to core damage frequency (CDF) and release category frequency. The licensee probed the results by performing sensitivity studies for operator actions, common cause failures (CCFs), ventilation dependencies, and performed importance analyses for top events, split fractions, and operator actions.

Consistent with this definition, the IPE identified seven "vulnerabilities" including AC power generation failure on station blackout (SBO), RCP seal LOCA on SBO, loss of emergency switchgear room HVAC, 4160 V bus fast transfer failures, PORV failure to reclose after loss of offsite power, battery capacity on SBO, and reactor trip breaker failure. The enhancements that the licensee is considering to address these vulnerabilities are discussed in Section II.7 of this report.

Based on the review of the BV-2 IPE submittal and associated documentation, the staff finds reasonable the licensee's IPE conclusion that no other fundamental weakness or severe accident vulnerabilities now exist at BV-2. The staff finds the BV-2 IPE process capable of identifying severe accident risk contributors (or vulnerabilities) and that such capability is consistent with the objective of Generic Letter 88-20.

2. Front-End Analysis

The staff examined the front-end analysis for completeness and consistency with accepted PRA practices.

The front-end IPE analysis used the large event tree/small fault tree or alternatively, modularized and linked event trees methodology for CDF quantification. This method involves direct modeling of the impact of the dependencies in the event trees. The front-end analysis consists of two modules, one for support systems and another for the front-line systems. The RISKMAN (Ref. 12) software links together both of the modules and eliminates the need for support states creating essentially a large event tree. Fault trees are used to quantify system failure values which are used as inputs to the event tree nodes. The RISKMAN software was used for quantification of the CDF.

Based on the IPE description and response to questions, the staff finds the employed methodology clearly described and justified for selection. The chosen methodology is consistent with methods identified in Generic Letter 88-20.

The licensee's process identified 46 initiating events for BV-2 which are categorized in three broad groups: (1) loss of reactor coolant inventory, (2) transients, and (3) common cause initiating events including loss of support systems and internal floods. Nine of these events are internal flooding events. The initiating events were arrived at through a combination of approaches; i.e., comparisons with lists from other PRAs, and review of BV-2 trip summaries (for actual plant experience) and failure mode and effect analysis (FMEA) of plant systems for plant-specific initiators. Anticipated transients without scram (ATWS) events are not defined as separate initiating events; however, they are addressed through the development of an event tree for all initiating events that are followed by a failure of reactor trip. The staff has compared the list of initiators with lists from other PRAs and NUREG-2300 (Ref. 13), reviewed the licensee's response to questions on initiating events concerning instrument air and very small LOCAs, and found them to be consistent.

Systemic event trees were developed for each unique IE group. The IPE submittal contained all front-line and support system event trees, and special trees including steam generator tube rupture (SGTR) and ATWS event trees. System success criteria were presented for each IE category. The licensee has stated that system success criteria are basically derived from the updated final safety analysis report (UFSAR); however, where the licensee found these criteria "unrealistically conservative," success criteria were developed by application of engineering judgement based on documented analysis and previous PRAs using similar success criteria. The success criteria presented are stated to be consistent with those for Surry Unit 1 in NUREG-4550, Vol. 3 (Ref. 14), and the applicability of these success criteria was established by review of the BV-2 UFSAR and plant-specific thermal-hydraulic calculations. However, the licensee noted certain exceptions to those criteria from Surry Unit 1 as follows:

- For bleed and feed cooling, one high head safety injection (HHSI) pump with one of three cold leg injection paths and <u>one of the three pressurizer PORVs</u> is adequate for heat removal.

- The ATWS success criteria developed for BV-2 is different from Surry Unit 1 in that the criteria developed for adequate pressure relief is adapted to BV-2 from the analysis provided in WCAP-11992 (Ref. 15). This criteria is in the top event description PA for the ATWS tree.

In general the staff finds the BV-2 event trees and special trees to be consistent with regard to initiating events, associated success criteria and dependencies between top events.

The IPE submittal explicitly addressed dependencies by providing dependency matrices which identified support to support and support to frontline systems dependencies on a "train" basis and by explicitly modeling dependency impacts in the event tree logic including service water, AC, DC, vital power, and emergency switchgear HVAC. A FMEA was performed for the HVAC systems at BV-2 to determine locations judged to be sensitive to HVAC failures. Two systems were found to be important; the diesel generator building HVAC, which is included in the emergency diesel system analysis and the emergency switchgear HVAC which is included as a top event in the support system event tree. It was determined that the emergency switchgear HVAC is a significant initiating event, contributing approximately 12% to CDF.

The BV-2 PRA Model was quantified with generic data from DLC's consultant's (PLG Inc.) database PLG-500 (Ref. 16), and plant- and unit-specific data obtained from plant records of operating experience where available. Plant-specific data was incorporated into the generic data through use of Bayesian updating techniques. BV-2 specific data for components was obtained through review of plant records of failures, maintenance and test records between November 1987 and December 1988. These records provided data for important components such as auxiliary feedwater (AFW) and emergency core cooling system (ECCS) pumps and electrical system equipment. The licensee has stated that a review of plant-specific data for initiating events was performed and that the development of the distributions is similar to the approach in PLG-500 (Ref. 16), except that initiating events involving system failures used plant-specific system analyses.

As indicated previously, the BV-2 IPE has considered impacts of common cause failures (CCFs) due to system dependencies by incorporating them explicitly in the event tree logic. Additional CCFs due to such conditions as design errors, construction errors, procedural deficiencies, and unforseen environmental conditions are accounted for in their contributions to system unavailabilities through plant-specific component CCF factors. The methodology used for quantification of CCF factors for the BV-2 submittal is the multiple Greek letter method; The staff notes that the licensee's analytic treatment of CCF is consistent with NUREG/CR-2300 (Ref. 13) and NUREG/CR-4780 (Ref. 17).

The internal flooding analysis described in the IPE represents a summary of the more detailed report (Internal Flood Analysis, Appendix E of BV-2 Probabilistic Risk Assessment, PLG-0730 (Ref. 7)). A screening analysis was performed for key safety equipment, locations to assess potential flood sources and flow paths, mitigating features (such as drains, detection, isolation) and system impacts to determine the potential for flooding as an initiating event. Specific maintenance actions considered for flooding were not described in the submittal; however, in response to questions on flooding, the licensee indicated that they were included in the initiating event database. A number of flooding events were postulated and combined with independent failures through the plant model to arrive at an estimated CDF due to flooding. The IPE reported a frequency of flood-induced CDF (7.3E-6/yr; 3.9% of the CDF) with a control building flood (CBFL) from service water contributing approximately 2% of the CDF. Based on the review of the description of the internal flood analysis provided in the submittal and the response to questions, the staff finds the IPE flood assessment to be consistent with Generic Letter 88-20.

The submittal identified the dominant accident sequences in accordance with the reporting guidelines in NUREG-1335 and presents the 100 highest frequency sequences. The IPE estimates the mean CDF as 1.9E-4/yr. Loss of offsite power contributes 14.8%, emergency switchgear HVAC contributes 12.2%, and small break LOCAs (isolable and non-isolable) and loss of a single train of 4160V AC power contribute 21.9% and 12.5% respectively, and SGTR contributes 3.7%. For sequences of specific interest, a large fraction of the CDF is associated with RCP seal LOCA (53.4%) (the licensee used the NUREG-1150 RCP seal LOCA model); a large portion of which is caused by SBO (25.3% CDF) and loss of switchgear ventilation. As expected, the largest system contributor to CDF is failure of high head safety injection (see Section 6.0), because of its dependency on electric power and service water.

In addition the submittal also provides a discussion of the top 12 highest frequency sequences which account for about 42% of the CDF with the single highest frequency sequence being complete loss of emergency switchgear HVAC responsible for about 11%. Each of the other sequences (not in the top 12) contributes less than 1.5%.

Based on the staff's review of the front-end analysis and the staff's finding that the employed analytical techniques are consistent with other NRC reviewed and accepted PRAs and capable of identifying potential core damage vulnerabilities, the staff finds the IPE front-end analysis meets the intent of Generic Letter 88-20.

3. Back-End Analysis

The staff examined the BV-2 back-end (Level 2) analysis for completeness and consistency with acceptable PRA practices. The analysis utilized methodology similar to that exercised in the Surry-NUREG-1150 PRA (Ref. 18), and employed Revision 14 and 16 of the MAAP-3.0B computer code (Ref. 19) and the CORCON-MOD2 computer code (Ref. 20) to model the containment thermal response. As part of the review, the staff examined the licensee's methodology, documentation of analytical codes exercised, and input data. The staff found the approach to be consistent with Generic Letter 88-20, Appendix 1 (Guidance on the Examination of Containment System Performance).

Sequences generated from the front-end (Level 1) analysis were grouped into plant damage states (PDSs) which are characterized by the thermodynamic conditions in the reactor coolant system (RCS) and containment, and availability of the plant systems and features. The PDSs were used as the entry states to the containment event trees (CETs). To develop the CETs for BV-2, the licensee reviewed each of the 71 top events identified in NUREG-1150 for the Surry accident progression event tree (APET) for applicability to the BV-2 CETs. 25 top events were selected as appropriate. The CET end states were subsequently binned into 21 release categories, and these 21 categories were further consolidated into 4 release category groups for simplification in reporting. The licensee first quantified the CETs for a number of key PDSs to check out the rules used for assigning CET split fractions and binning sequences to appropriate release categories. The licensee later used the RISKMAN software to link the CETs to the Level 1 event trees and generated the frequencies of the release categories. The release category source terms for BV-2 were developed using the existing Surry source terms.

The licensee noted that both BV-2 and Surry Unit 1 are three-loop, Westinghouse PWRs with large dry subatmospheric containment structures using reinforced concrete with steel liners, and were designed and constructed by Stone and Webster for a design pressure of 45 psig. Because of the similarity between the two plants, the licensee compared the design of the BV-2 containment with that of the Surry Unit 1 containment. The comparison included containment geometry, material properties, rebar quantities and pattern, liner thickness, hatches and penetration configurations, and calculated and actual pressure test data. The comparison indicated that the containment failure distributions for Surry Unit 1 would bound the BV-2.

The IPE submittal estimated the following conditional containment failure probabilities:

Large, Early Containment Failure and Bypasses	-	0.05
Small, Early Containment Failure and Bypasses	-	0.26
Late Containment Failures	-	0.45
No Containment Failure	-	0.24

The licensee noted that high pressure melt ejection is the major contributor to early containment failures. Therefore the licensee performed three sensitivity analyses to determine the impact on the containment releases due to operator actions to arrest core damage before vessel breach, temperature-induced RCS failures prior to vessel breach, and deliberate depressurization of the RCS after core damage occurs. The results of the sensitivity analyses indicated that containment releases can be substantially reduced if the IPE analysis accounts for these considerations. The staff notes that the IPE results have not taken credit for these considerations, and that the licensee plans to pursue some of these considerations during accident management program development as noted below.

The licensee took minimal credit for recovery actions following core damage in its back-end analysis. However, the licensee identified the changes to plant procedures and training that will enhance the operator's response to SGTR sequences. Procedures for the following operator actions needed during SGTR sequences are being considered:

(1) Depressurize the intact steam generators during a SGTR event in which HHSI fails.
(2) Close and gag a stuck-open steam generator safety valve during a SGTR event. (Although no credit had been taken in the IPE analysis, this option is being considered by the licensee).

The staff noted that the use of the licensee's IPE to identify the above recovery actions is consistent with the intent of Generic Letter 88-20.

In addition, the licensee plans to evaluate the following items further during accident management program development:

(1) Use diesel-driven fire pump to add water to the reactor cavity.
(2) Depressurize the primary system in SBO sequences.
(3) Throttle the quench spray pumps to conserve RWST water for core injection.
(4) Improve the guidance to the operators regarding the "LOCA Outside Containment" procedures so that the operators would be knowledgeable of key isolation valves.

Although the review team did not examine closely the merits of these items in detail, the staff notes that the licensee is applying PRA/IPE findings to enhance plant safety. The staff finds the licensee's actions reasonable.

The licensee employed an adequate process to understand and quantify severe accident progression. The process of determination of conditional containment failure probabilities and containment failure modes was consistent with the intent of Generic Letter 88-20, Appendix 1. Dominant contributors to containment failure were found to be consistent with insights from other PRAs of plants of similar design. The IPE characterized containment performance for each of the CET end-states by assessing containment loading. The licensee's IPE addresses the most important severe accident phenomena normally associated with large dry containments, that is, direct containment heating (DCH), Induced Steam Generator Tube Rupture (ISGTR) and hydrogen combustion. The licensee considered failure of non-metallic seals used for containment penetrations. In addition, the licensee used a fault tree analysis to model containment isolation failures. The staff's review did not identify any obvious or significant problems or errors in the back-end analysis. The overall assessment of the back-end analysis is that the licensee has made reasonable use of PRA techniques in performing the back-end analysis, and that the techniques employed are capable of identifying severe accident vulnerabilities. Based on these findings, the staff concludes that the licensee's back-end IPE process is consistent with the intent of Generic Letter 88-20.

4. Human Factor Considerations

The IPE submittal is essentially complete with respect to the type of information and level of detail requested in NUREG-1335.

The Beaver Valley 2 IPE submittal provides thorough documentation of the human reliability analysis conducted. Three general types of human actions were evaluated as part of the IPE, including those that occur during maintenance and operations prior to an initiator, those that are an integral part of plant response to an initiating event, and those actions that involve recovery from unexpected failures that completely or partially disable automatic system response during a plant transient. This human action taxonomy is logical and representative of those used in other PRAs, and it supports the identification of important human actions.

The HRA methodology employed in the IPE utilized the approach described in NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications" (Ref. 21), and an adaptation (the failure likelihood index) of the success likelihood index methodology (SLIM) to quantify the human actions in the plant response model of the PRA. This methodology considers human performance shaping factors for each dynamic human action and associated human error rate (HER) mean value and, using the RISKMAN software, the associated uncertainty distribution.

The IPE submittal provides a discussion of the most likely accident sequences and includes a description of the important human actions in each sequence.

The results of an analysis of the importance of human actions to core damage is provided in the submittal and is repeated in the appendix to this staff evaluation report.

The licensee's IPE submittal also includes a sensitivity study to determine potentially important human actions for which substantial credit was taken in the base case quantification. The base case plant models were rerun after setting all of the human actions evaluated to the base case error rate or 0.1, whichever was higher. The IPE submittal includes a discussion of the new sequences, and associated human actions, that appeared above 1.0E-7 in the sensitivity study.

In summary, and based on a review of the licensee's IPE submittal, the staff finds the licensee's assessment of human reliability, conducted as part of the BV-2 IPE, capable of discovering severe accident vulnerabilities from human errors consistent with the intent of Generic Letter 88-20. The HRA methodology described in the licensee's IPE submittal supports the quantitative understanding of the overall probability of core damage during plant operations, as well as an understanding of the contribution of human actions to that probability. Human-related plant improvements that are planned or under review, such as those to implement procedures and training, are expected to enhance human reliability and plant safety. In addition, the licensee's stated intention to maintain the PRA will ensure that a mechanism exists for the licensee to continue to identify and evaluate the risk significance of potentially important human actions during plant operation and maintenance.

5. Containment Performance Improvements (CPI)

Generic letter 88-20, Supplement 3 (Ref. 6), contains CPI recommendations which focus on the vulnerability of containments to severe accident challenges. For large dry containments, such as the BV-2 design, the reference contains a recommendation that IPEs consider hydrogen production and control during severe accidents, particularly the potential for local hydrogen detonation.

Containment failure due to containment overpressurization from global hydrogen combustion has been addressed explicitly by the licensee in the BV-2 IPE. Based on the peak containment pressures determined using the adiabatic burn assumption, the licensee found that deflagration is not likely to fail the BV-2 containment.

As a result of the evaluation and analysis of the BV-2 containment design and comparison to the Surry containment design, the licensee does not find hydrogen "pocketing" inside the containment building to be of concern. The licensee bases this conclusion upon the following observations:

(1) During the development of the MAAP parameter file, it was noted that a lot of junction areas existed between compartments relative to the potential hydrogen release points, which would allow dispersion of hydrogen.

(2) The containment walkdown confirmed the above finding.

(3) The MAAP analyses indicated that the hydrogen was well mixed, especially when the containment sprays were functioning.

The licensee's conclusion on hydrogen "pocketing" is also consistent with NUREG-1150 for both Surry and Zion (plants with large dry containments).

The staff, therefore, concludes that the licensee's response to CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20 and associated Supplement 3.

6. DHR Evaluation

In accordance with the resolution of USI A-45, the licensee performed an examination of BV-2 to identify DHR vulnerabilities. The results of the IPE provide indications of the importance of the systems that provide the DHR function as a response to the initiating events postulated in the IPE.

The following system features were considered in the DHR evaluation:

- Main feedwater (MFW)
- AFW
- Bleed and feed cooling
- Steam generator depressurization to cool down the RCS
- Residual heat removal (RHR)

The contribution to CDF from the DHR systems with all support systems available as identified in the submittal is small, e.g., contribution from the MFW is 0.1 %, contribution from RHR is less than 0.1 %. In response to the staff's request for additional information, the licensee provided information on the contribution from the failure of DHR systems due to electric power and non-electric power support system failures. The largest contributor due to loss of electric power is the failure of HHSI, involving 45.6% of the core damage sequences. The largest contributor due to loss of non-electric power support system is the failure of HHSI, involving 16.4% of the core damage sequences due to loss of both service water headers to the HHSI/charging pump coolers. Additionally the licensee provided information on the worth of the capability to use feed and bleed (the CDF increased from 1.9E-4 to 2.2E-4, or about 3E-5, if the capability was unavailable) and of the capability to realign the electric motor driven MFW pumps (the CDF increased from 1.9E-4 to 2.3E-4, or about 2.3E-5, if the capability was unavailable).

Based on the process that the licensee used to search for DHR vulnerabilities, and review of plant-specific features, the staff finds the licensee's DHR evaluation to be consistent with the intent of Generic Letter 88-20, and resolution of USI A-45.

7. <u>Licensee Actions and Commitments from the IPE</u>

As part of the IPE process, the licensee identified seven "vulnerabilities" listed as follows:

(1) AC power generation failure on SBO,
(2) RCP seal LOCA on SBO,
(3) Loss of emergency switchgear HVAC,
(4) 4160 V bus fast transfer failures,
(5) PORV failure to reclose after loss of offsite power,
(6) Battery capacity on SBO.
(7) Reactor trip breaker failure.

The licensee addressed these vulnerabilities by identifying the following list of items for possible implementation:

(1) Provide Beaver Valley Units 1 and 2 with a 4160V bus crosstie capability.
(2) Incorporate modifications to reduce the potential for RCP seal LOCA.
(3) Review alarm response procedures for loss of emergency switchgear HVAC.
(4) Prepare explicit procedure for and provide training on repair of breaker on failure of 4160V bus fast transfer.
(5) Eliminate challenges to the PORV by defeating the 100% load rejection capability.
(6) Enhance procedures on shedding loads or using portable battery chargers for loss of all AC power scenarios.
(7) Provide the capability for the operators to remove power to the control rods in the event of a failure of automatic reactor trip.

The licensee plans to implement the first item (planned in response to the SBO rule). The modification is to provide Beaver Valley Units 1 and 2 with a 4160V bus crosstie to allow SBO cooling capability by using the emergency diesel generators of the other unit as an alternate AC source. The other six items are still under review.

As noted in Section II.3, the licensee is also in the process of evaluating new procedures for the following operator actions to deal with SGTR events:

(1) Depressurize the intact steam generators during a SGTR event in which HHSI fails.
(2) Close and gag a stuck-open steam generator safety valve during a SGTR event.

The licensee recognizes the potential benefits of a PRA and plans to use and maintain it. In addition, as noted in Section II.3, the licensee plans to evaluate the following items further during accident management program development:

(1) Use diesel-driven fire pump to add water to the reactor cavity.
(2) Depressurize the primary system in SBO sequences.
(3) Throttle the quench spray pumps to conserve RWST water for core injection.

(4) Improve the guidance to the operators regarding the "LOCA Outside Containment" procedures so that the operators would know which key isolation valves need to be closed.

Although the review team did not examine closely the merits of these items in detail, the staff notes that the licensee is applying PRA/IPF findings to enhance plant safety. The staff finds the licensee's actions reasonable. The staff believes the licensee's proposed actions in response to the IPE identified "vulnerabilities" would be consistent with the intent of Generic Letter 88-20.

III. CONCLUSION

The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal and the associated supporting information, the staff finds reasonable the licensee's IPE conclusion that, except for the identified "vulnerabilities," no other fundamental weakness or severe accident vulnerabilities exist at BV-2. The staff notes that:

(1) DLC personnel were involved in the development and application of PRA techniques to the BV-2 facility, and that the associated walkdowns and documentation reviews constituted a viable process for confirming that the IPE represents the as-built, as-operated plant.

(2) The front-end IPE analysis appears complete, with the level of detail consistent with the information requested in NUREG-1335. In addition, the employed analytical techniques are consistent with other NRC reviewed and accepted PRAs and capable of identifying potential core damage vulnerabilities.

(3) The back-end analysis addressed the most important severe accident phenomena normally associated with large dry containments. The techniques employed in the back-end analysis are capable of identifying severe accident vulnerabilities. No obvious or significant problems or errors were identified.

(4) The HRA allowed the licensee to develop a quantitative understanding of the contribution of human errors to CDF and containment failure probabilities. The assessment of human reliability was capable of discovering severe accident vulnerabilities from human errors.

(5) Based on the licensee's IPE process used to search for DHR vulnerabilities, and review of BV-2 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.

(6) The licensee's response to CPI Program recommendations, which include searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of

Generic Letter 88-20 Supplement 3.

In addition, and consistent with the intent of Generic Letter 88-20, the staff believes the licensee's peer review process provided assurance that the IPE analytic techniques had been correctly applied and that the effort had been properly documented.

Based on the above findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the BV-2 facility, has gained a quantitative understanding of core damage and fission product release, and responded appropriately to safety improvement opportunities. The staff, therefore, finds the BV-2 IPE process acceptable in meeting the intent of Generic Letter 88-20. The staff also notes that the licensee's intent to continue use and maintain its PRA document will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

IV. REFERENCES

1. NRC letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter No. 88-20, dated November 23, 1988.

2. NUREG-1335, "Individual Plant Examination: Submittal Guidance - Final Report," USNRC, August 1989.

3. J. Sieber of Duquesne Light Company to USNRC, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73 Generic Letter 88-20 (TAC No. M74378)," March 17, 1992.

4. Duquesne Light Company, "Probabilistic Risk Assessment for the Individual Plant Examination Final Report BV-2," August 1991.

5. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," - Generic Letter No. 88-20, Supplement No. 1, dated August 29, 1989.

6. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities except Licensees for Boiling Water Reactors with MARK I Containments, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20, Supplement No. 3, dated July 6, 1990.

7. PLG-0730, "Beaver Valley Unit 2 Probabilistic Risk Assessment," Pickard, Lowe and Garrick, Inc., and Stone and Webster Engineering Corporation, December 1989.

8. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4), dated June 28, 1991.

9. A. De Agazio of USNRC to J. Sieber of Duquesne Light Company, "Generic Letter 88-20 Individual Plant Examination for Severe Accident Vulnerabilities - Request for Additional Information (TAC No. M743779)," July 15, 1992.

10. J. Sieber of Duquesne Light Company to USNRC, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73 Generic Letter 88-20 (TAC No. M74378)," September 11, 1992.

11. J. Sieber of Duquesne Light Company to USNRC, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73 Generic Letter

88-20 (TAC No. M74378)," October 26, 1992.

12. "RISKMAN PRA Workstation Software, User Manual II: Systems Analysis," Release 3.0, Pickard, Lowe and Garrick, Inc., November 1989.

13. NUREG/CR-2300, "PRA Procedures Guide," January 1983.

14. NUREG/CR-4550, Vol. 3, Rev. 1, "Analysis of CDF From Internal Events: Surry Unit 1," 1990.

15. WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance with ATWS Rule Basis for Westinghouse PWR's," December 1988.

16. PLG-500, "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," Pickard, Lowe and Garrick, Inc., September 1990.

17. NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," January 1988.

18. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," June, 1989.

19. "Modular Accident Analysis Program (MAAP) 3.0 B Users Manual," prepared for Electric Power Research Institute by Fauske and Associates, Inc., July 16, 1990.

20. NUREG/CR-3920, "CORCON-MOD2: A Computer Program for Analysis of Molten Core Concrete Interactions," August 1984.

21. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983.

APPENDIX
Beaver Valley Unit 2 DATA SUMMARY SHEET*
(INTERNAL EVENTS)

- Total core damage frequency (CDF): 1.9E-4/year

- Major initiating events and contribution to CDF:

	Contribution
Loss of offsite power (LOOP)	14.8%
Loss of switchgear ventilation	12.2%
Small LOCAs (isolable)	11.2%
Small LOCAs (non-isolable)	10.7%
Loss of AC power (orange)	7.7%
Loss of AC power (purple)	4.8%
Loss of vital bus (channel 1 or 2)	7.6%
Steam generator tube rupture (SGTR)	3.7%

(The highest ranked sequence is a loss of emergency switchgear HVAC, and three of the top nine ranked sequences are SBO sequences initiated by LOOP.)

- Major contributions to dominant core damage sequences:

  Reactor coolant pump (RCP) seal LOCAs caused by station blackout (SBO) stemming from LOOP or loss of emergency switchgear room heating, ventilation and air conditioning (HVAC). Initiaters lead to loss of RCP seal cooling and high head safety injection (HHSI).

  Small isolable LOCA with subsequent loss of both service water headers resulting in RCP seal LOCA and failure of HHSI and recirculating spray system.

  Small non-isolable LOCA with loss of HHSI suction from refueling water storage tank (RWST) and loss of cold leg injection path.

  Overall largest system contributor to CDF is the failure of HHSI, due to the system dependence on either electric power or service water.

- Major operator action failures:

  Operator fails to prevent HHSI suction path swapover from volume control tank (VCT) to RWST in sequences where RWST suction path is unavailable.

  Operators fail to recover offsite power.

  Operator fails to align filtered water supply to station air compressors following a loss of the primary closed cooling water system to prevent RCP seal degradation.

Operator prematurely secures safety injection.

Operator fails to recover emergency switchgear HVAC.

o Conditional containment failure probability given core damage:

Large, Early Containment Failure and Bypasses	-	0.05
Small, Early Containment Failure and Bypasses	-	0.26
Late Containment Failures	-	0.45
No Containment Failure	-	0.24

o Significant PRA findings:

Vulnerabilities Identified by the Licensee	Importance (% ofCDF)
(1) AC power generation failure on SBO	25.3
(2) RCP seal LOCA on SBO	18.8
(3) Loss of emergency switchgear HVAC	17.1
(4) 4160 V bus fast transfer failures	8.0
(5) Power operated relief valve (PORV) fails to reseat on LOOP	7.2
(6) Battery capacity on SBO	6.8
(7) Reactor trip breaker failure	4.2

The RCP seal injection and thermal barrier cooling are not both dependent on the primary closed cooling water system. However, they are both supported by service water.

HHSI pumps can be used for feed and bleed with one of the three PORVs.

Loss of containment instrument air will close the air-operated valves which provide cooling water to the RCP seal, bearings, and motors.

To prevent containment bypass in SBO sequences, operators needed to manually isolate the motor-operated valves.

For small LOCAs, operator actions are needed to provide makeup to RWST if containment sump recirculation fails.

o Plant and procedure changes based on PRA considerations:

Cross-tie Beaver Valley Units 1 and 2 with a 4160V bus to use the emergency diesel generators of the other unit.

Depressurize the intact steam generators during a SGTR event in which HHSI fails.

Close and gag a stuck-open steam generator safety valve during a SGTR event.

o Future potential improvements under evaluation:

Incorporate modifications to reduce the frequency of RCP seal LOCA.

Review alarm response procedures for loss of emergency switchgear HVAC to provide prompt operator actions.

Prepare explicit procedure and provide training on repair of breaker on failure of 4160V bus fast transfer.

Defeat the 100% load rejection capability to avoid PORV fr[illegible]cking open on demands.

Enhance procedures on shedding loads or using portable battery chargers for loss of all AC power scenarios.

Add the capability for the operators to remove power to the control rods in the event of reactor trip breaker failure.

Use diesel-driven fire pump to add water to the reactor cavity.

Depressurize the primary system in SBO sequences.

Throttle the quench spray pumps to conserve RWST water for core injection.

Improve the guidance to the operators regarding the "LOCA Outside Containment" procedures so that the operators would know which key isolation valves need to be closed.

o Future Activities: Periodic update of PRA

(* Information has been taken from the Beaver Valley Unit 2 IPE and has not been validated by the NRC staff.)