Final Safety Evaluation by
the Office of Nuclear Reactor Regulation
Related to Operation of
Beaver Valley Power Station Unit No. 1

6

POST-ACCIDENT SAMPLING SYSTEM
NUREG-0737, II.B.3

Introduction

The post-accident sampling system (PASS) is evaluated for compliance with the criteria in NUREG-0737, Item II.B.3. The licensee should provide information on the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g. noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737, II.B.3.

To satisfy the requirements, the licensee should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary to comply with NUREG-0737, Item II.B.3, and (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements are met.

Evaluation

By letter dated August 31, 1982, April 15, 1983, June 5, 1984 and July 16, 1984 the licensee provided information on the PASS. Our evaluation is as follows:

Criterion: (1)

The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

The PASS has sampling and analysis capability to promptly obtain and analyze reactor coolant samples and containment atmosphere samples within three hours from the time a decision is made to take a sample. The PASS was not designed to have a backup power source. Sample collection and analysis will not be possible if offsite power is lost. However, a heavily shielded backup laboratory and counting facilities are maintained in the Emergency Response Facility (ERF) for post-accident sampling and analysis in the event that the hot chemistry laboratory would be unavailable or inaccessible. Furthermore, the ERF will have a backup diesel generator power supply. We determined that these provisions meet Criterion (1) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (2)

The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes);

b) hydrogen levels in the containment atmosphere;

c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

d) Alternatively, have in-line monitoring capabilities to perform all or part of the above analyses.

The PASS is capable of obtaining a grab sample of the containment atmosphere, the reactor coolant system or the containment sump. Radionuclides analysis of grab samples can be done in the chemistry hot laboratory or the ERF lab. In-line monitors are also available for hydrogen analysis of containment atmosphere (Item II.F.1.6). The boron, chloride and pH levels will be analyzed by in-line probes. Grab sample capability is available should the electrode system fail.

The licensee provided a procedure for estimating the degree of reactor core damage based on the Westinghouse Owners Group generic methodology, Revision 1, dated March 1984, which relates to post-accident core damage with measurements of radionuclide concentrations in the reactor coolant and containment atmosphere.

The procedure takes into consideration other physcial parameters such as reactor core temperature data, reactor water level, sample location, and containment radiation levels and hydrogen concentrations. We determined that these provisions meet Criterion (2) and are, therefore, acceptable.

Criterion: (3)

> Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Reactor coolant and containment atmosphere sampling during post accident conditions does not require an isolated auxiliary sysem to be placed in operation in order to perform the sampling function. The PASS provides the ability to obtain samples from each reactor coolant hot leg, the residual heat removal system (RHR), the containment sump, and the containment atmosphere without using an isolated auxiliary system. The licensee's response to Criterion (3) is acceptable since PASS sampling is performed without requiring operation of an isolated auxiliary system and PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they need to operate.

Criterion: (4)

> Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Hydrogen concentrations as low as 3 cc/kg can be measured. Dissolved oxygen is indicated directly in ppm as it is read by the in-line analyzer. On the 0-1 ppm scale, oxygen concentration as low as 0.02 ppm can be measured. Furthermore, hydrogen and oxygen concentrations can be measured in the laboratory using a gas chromatograph. We determined that these provisions meet Criterion (4) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (5)

> The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licesee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

An in-line chloride analyzer is provided which meets the 96-hour chloride limit for a fresh water plant. Additionally, grap samples will be available for laboratory analysis within four days. We determined that these provisions meet Criterion (5) and are, therefore, acceptable.

Criterion: (6)

> The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

The licensee has performed a shielding analysis to ensure that operator exposure while obtaining and analyzing a PASS sample is within acceptable limits. This operator exposure includes entering and exiting the sample panel area, operating sample panel manual valves, positioning the grab sample into the shielded transfer carts, and performing manual sample dilutions, if required, for isotopic analysis. PASS personnel radiation exposures from reactor coolant and containment atmosphere sampling and analysis are within 5 rem whole body and 75 rem extremities which meet the requirements of GDC (19) and Criterion (6) and are, therefore, acceptable.

Criterion: (7)

The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Boron analysis of the reactor coolant will be performed by in-line boron analyzer with a measurement capability from 0 ppm to 6,000 ppm under accident conditions. Prior to time when the boron analyzer is operational, boron can also be analyzed using diluted reactor coolant sample. We find that this provision meets the recommendations of Regulatory Guide 1.97, Rev. 2 and Criterion (7) and is, therefore, acceptable.

Criterion: (8)

If in-line monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

An in-line chemical analysis panel is provided for reactor coolant pH, boron, oxygen and hydrogen concentrations. Also, a backup (diluted and undiluted) reactor coolant grab sample can be obtained for the offsite analysis. We find that these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion: (9)

The licensee's radiological and chemical sample analysis capability shall include provisions to:

a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source term given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1μ Ci/g to 10 Ci/g.

b) Restrict background levels or radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. Radioisotope analysis can be performed in the chemistry laboratory in the station or in the Emergency Response Facility lab. Radiation background levels will be restricted by shielding and ventilation in the radiological and chemical analysis facilities such that analytical results can be obrained within an acceptably small error (approximately a factor of 2). We find that these provisions meet Criterion (9) and are, therefore, acceptable.

Criterion (10):

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Analytical accuracies are estimated for gamma spectra, boron, chloride, total dissolved gases, and pH based on normal test solutions. The systems performance will be verified when the PASS is installed.

All instruments were purchased with certification that they would function in a radiation field exceeding 10^4 rads/gram of reactor coolant. Boron, pH, and chloride analyzer will be calibrated on a six month frequency while dissolved hydrogen and oxygen analyzers will be calibrated every 18 months. The operators will be retrained on a semi-annual basis.

The analytical accuracies were provided to describe radiological and chemical status of the reactor coolant system. The licensee also provided information on the measurement ranges and sensitivity of the procedure to demonstrate, on the standard test matrix, that the selected procedures and instrumentation achieved acceptable accuracies. We determined that these provisions meet Criterion (10) and are, therefore, acceptable.

Criterion: (11)

In the design of the post-accident sampling and analysis capability, consideration should be given to the following items:

a) Provisions for purging sample lines, for reducing plateout in sample line, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-effciency particulate air (HEPA) filters.

The licensee has addressed provisions for purging to ensure samples are representative, size of sample line to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. Information was also provided regarding containment atmosphere sample line heat tracing to limit iodine plateout.

We determined that the licensee meets Criterion (11) of Item II.B.3 of NUREG-0737.

Conclusion

On the basis of our evaluation, we conclude that the post-accident sampling system meets all eleven criteria of Item II.B.3 in NUREG-0737, and is, therefore, acceptable.

Principal Contributors

P. Wu, Reviewer

P. Tam, Project Manager

Dated: September, 1984

September 6, 1984

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Docket Nos. 50-250
and 50-251

Mr. C. W. Williams, Jr., Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

SUBJECT: PROPOSED SPENT FUEL POOL EXPANSION FOR TURKEY POINT UNITS 3 AND 4 - REQUEST FOR ADDITIONAL INFORMATION

By letter dated March 14, 1983, you requested that the Technical Specifications for Turkey Point Plant Units 3 and 4 be modified to expand the spent fuel storage facilities. The expansion is necessary to accommodate an expected increase in the inventory of spent fuel assemblies above the capacity of the existing storage facilities.

The staff is currently reviewing your initial request and the additional information provided by letter dated July 2, 1984. The staff needs the additional information identified in the enclosure to this letter. The request is related to decay heat loads, testing and inspections of the storage racks, cooling water flow and load handling during rerack operations. This request will be discussed at a meeting to be held in Bethesda, Maryland, on September 10, 1984. We request your response as soon as practicable in order to meet our review schedule.

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

/s/SVarga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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DMcDonald/ts SVarga
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PDR ADOCK 05000250

5

REQUEST FOR ADDITIONAL INFORMATION
SPENT FUEL STORAGE CAPACITY EXPANSION - TURKEY POINT UNITS 3 AND 4

5. We have performed a spent fuel decay heat load calculation in accordance with the Standard Review Plan Section 9.1.3 and Branch Technical Position ASB 9-2 which does not agree with your calculation for the normal heat load conditions. Provide the results of a revised decay heat load analysis using the equations in the above referenced documents. Provide the results of the decay heat load analysis for the abnormal heat load case (one full core offload with the balance of the pool filled with half core refuelings). Based on these two analyses, provide a revised response to Question No. 1 which was transmitted to you on May 11, 1984.

6. The updated FSAR indicates that there is only one 7.96 MBTU/hr spent fuel pool heat exchanger. This is clearly undersized as your analysis indicates a 8.82 MBTU/hr heat load for the existing racks. Provide a commitment to install a second full capacity heat exchanger by the next refueling outage.

7. The updated FSAR is not clear. Either 1) verify that there is an interconnection between the spent fuel pool and the RHR system and provide P&ID(s) which show the interconnection, 2) commit to provide the interconnection in (1) by the next refueling or 3) provide the results of an analysis which shows that no offsite dose limits and personnel exposure limits will be exceeded by allowing the pool to boil with makeup from only the seismic Category I source(s).

8. The updated FSAR indicates that the spent fuel pool cooling system is designed for a maximum temperature of 200°F and the storage capacity submittal indicates that the spent fuel pool is designed for a temperature of 150°F. Provide a discussion of the effects of a sustained pool water temperature of 212°F on the pool and on the cooling system. Provide the anticipated time until failure of the pool structure and the effects of the anticipated failure.

9. The submittal is unclear as to the intended use of the new fuel storage facility. Is it your intention to convert the new fuel storage facility for storage of spent fuel? Provide a discussion of your intended use of the new fuel storage facility and any changes between the existing system and the proposed system.

10. The submittal stated that the temporary crane, racks, and staging platform will have to be carried over the exclusion area identified in the drawings submitted in response to NUREG-0612. Therefore, provide the safe load path drawings requested in our Question No. 4.

11. Verify that the procedures will require that the transportation of loads follow the safe load paths identified on the drawings that you will provide in response to Question Number 10.

12. Will any special lifting devices be used? For each special lifting device, provide a comparison to Guideline 4 of NUREG-0612, "Special Lifting Devices," and verify that it is single failure proof.

13. ~~The new~~ racks will hold more spent fuel than the existing racks, therefore it is not clear that a cask drop accident with the new racks will be bounded by a cask drop accident with the old racks. Provide a discussion of the cask drop accident with the new racks.

J. O'REILLY

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

August 28, 1984

84 SEP 5 A10:19

Docket Nos. 50-280
and 50-281

Mr. W. L. Stewart, Vice President
Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

We have reviewed your July 6, 1982, response to our May 21, 1982, letter related to NRC IE Bulletin 80-11 (Masonry Wall Design) for the Surry Power Station. We find that we need the information identified in the enclosure.

Also, it is our understanding that you have replaced portions of block walls by metal siding around the spent fuel pool. We request that you summarize your activities regarding the block walls around the spent fuel pool and provide the basis for which these actions satisfy requirements of IE Bulletin 80-11.

Please provide your response within 45 days of receipt of this letter.

The reporting and/or recordkeeping recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore OMB clearance is not required under P.L. 96-511.

Sincerely,

Steven A. Varga

Steven A. Varga, Branch Chief
Operating Reactors Branch #1
Division of Licensing, NRR

Enclosure:
As stated

cc w/enclosure:
See next page

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PDR 50-280 Q 840828

4.

Mr. W. L. Stewart
Virginia Electric and Power Company

Surry Power Station
Units 1 and 2

cc: Mr. Michael W. Maupin
Hunton and Williams
Post Office Box 1535
Richmond, Virginia 23213

Mr. J. L. Wilson, Manager
Post Office Box 315
Surry, Virginia 23883

Donald J. Burke, Resident Inspector
Surry Power Station
U.S. Nuclear Regulatory Commission
Post Office Box 166, Route 1
Surry, Virginia 23883

Mr. Sherlock Holmes, Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

W. T. Lough
Virginia Corporation Commission
Division of Energy Regulation
Post Office Box 1197
Richmond, Virginia 23209

Regional Radiation Representative
EPA Region III
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

James P. O'Reilly
Regional Administrator - Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

MASONRY WALL DESIGN
IE BULLETIN 80-11
REQUEST FOR ADDITIONAL INFORMATION
SURRY UNITS 1 AND 2

With respect to the boundary conditions used in the analysis, the licensee indicated in Reference 1 that fixity was assumed at the base of a block wall built on a concrete slab. Also, at the perpendicular intersection of two block walls, fixity has been assumed in the corner joints formed by the alternating courses of running bond. The licensee is requested to provide the technical basis for assuming fixed-end conditions for these cases. It is believed that without some clamping devices to prevent rotation at the wall boundary, the assumed boundary conditions may not be valid.

REFERENCES

1. R. H. Leasburg (Virginia Electric and Power Company) Letter with Attachments to Director of Nuclear Reactor Regulation (NRC). Subject: Masonry Wall Design (IE Bulletin 80-11), Surry Power Station Units 1 and 2. July 6, 1982.

4

MASONRY WALL DESIGN
IE BULLETIN 80-11
REQUEST FOR ADDITIONAL INFORMATION
SURRY UNITS 1 AND 2

With respect to the boundary conditions used in the analysis, the licensee indicated in Reference 1 that fixity was assumed at the base of a block wall built on a concrete slab. Also, at the perpendicular intersection of two block walls, fixity has been assumed in the corner joints formed by the alternating courses of running bond. The licensee is requested to provide the technical basis for assuming fixed-end conditions for these cases. It is believed that without some clamping devices to prevent rotation at the wall boundary, the assumed boundary conditions may not be valid.

REFERENCES

1. R. H. Leasburg (Virginia Electric and Power Company) Letter with Attachments to Director of Nuclear Reactor Regulation (NRC). Subject: Masonry Wall Design (IE Bulletin 80-11), Surry Power Station Units 1 and 2. July 6, 1982.

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

August 17, 1984

3

Docket No. 50-373

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING SRV BLOWDOWN TEST

In the La Salle Safety Evaluation Report (SER), the NRC staff indicated that you had committed to perform a comprehensive safety relief valve (SRV) in-plant test to demonstrate that the calculated maximum local pool temperature of 200°F will not be exceeded. The 200°F local pool temperature limit analyses was based on NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria."

Since the issuance of the SER, the Mark II Owners Group, which Commonwealth Edison is a member, proposed alternative suppression pool limits. The alternative limits which are applicable to La Salle are contained in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," which supersede the criteria contained in NUREG-0487. The alternative local pool temperature limits set forth in NUREG-0783 are dependent on the steam mass flux and the amount of subcooling of the suppression pool water near the steam quench front. For La Salle, the new temperature limits range between 200°F and 216.5°F.

The staff and its contractor, the Brookhaven National Laboratory, have completed their evaluation of your report entitled, "La Salle County Station, Unit 1, In-Plant SRV Test, Evaluation of Suppression Pool Temperature Measurements." In this report, you present results to show that the average local-to-bulk pool temperature difference is 8.1°F. The corresponding 95/95 confidence level non-exceedance temperature is 12°F. These test results are also intended to confirm the adequacy of the suppression pool temperature monitoring system for providing a conservative measure of the bulk pool temperature.

The staff's evaluation concludes that the test report does not provide sufficient, pertinent data needed to permit the staff to determine a local-to-bulk temperature difference value suitable for use in the La Salle Mark II plant transient analyses. We base this conclusion on the fact that you did not satisfy the criteria set forth in NUREG-0487 and NUREG-0783. The number of sensors reported is insufficient to provide an acceptable spatial average. Also, a non-conservative bias could be introduced by the use of sensors T2 and T4, which are located on the basemat, 4½ feet below the quencher elevation and about 2 feet upstream of the quencher center relative to the bulk pool motion.

3.

Despite these deficiencies, it is still possible that the reported temperatures do provide a reasonable measure of local pool temperature. In order for the staff and its consultant to make this determination, the enclosed additional information is requested. If you have any questions regarding this matter, please contact A. Bournia, Project Manager.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Request for Add'l Info.

cc: See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
LA SALLE COUNTY STATION, UNIT 1

1. Provide the temperature histories recorded by all (32) operating temperature sensors during each of the seven extended blowdown tests. These data can be supplied in the same graphical format used in the test report (Figure C-1).

2. Identify where temperature sensors T32 and T33 are located relative to the end cap holes.

La Salle

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

cc: Philip P. Steptoe, Esquire
Suite 4200
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
188 West Randolph Street
Suite 2315
Chicago, Illinois 60601

Michael J. Jordan, Resident Inspector
La Salle, NPS, U.S.N.R.C.
P.O. Box 224
Marseilles, Illinois 61364

Chairman
La Salle County Board of Supervisors
La Salle County Courthouse
Ottawa, Illinois 61350

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

Department of Public Health
535 West Jefferson
Springfield, Illinois 62761
ATTN: Chief, Division of Nuclear Safety

The Honorable Tom Corcoran
United States House of Representatives
Washington, D.C. 20515

Chairman
Illinois Commerce Commission
Leland Building
527 East Capitol Avenue
Springfield, Illinois 62706

Mr. Gary N. Wright, Manager
Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

September 13, 1984

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Docket No. 50-334

Mr. J. J. Carey, Vice President
Duquesne Light Company
Nuclear Division
Post Office Box 4
Shippingport, PA 15077

Dear Mr. Carey:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING SALEM ATWS EVENT ITEMS 4.2.1 AND 4.2.2

In our continuing efforts to resolve the subject issues, we have developed a number of questions on the information you provided in your November 4, 1983 letter (See enclosure).

We request that you respond to these questions within 45 days of receipt of this letter. If you have difficulty meeting the target date, or you need clarification on any of the questions, please feel free to contact your project manager, Mr. Peter Tam.

Sincerely,

/s/SVarga

Steven A. Varga, Branch Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

BEAVER VALLEY 1

Duquesne Light Co., the licensee for the Beaver Valley Unit 1 Power Station submitted their response[1] to Generic Letter 83-28 on November 4, 1983. The submittal has been reviewed with respect to items 4.2.1 and 4.2.2 of the Generic Letter. The following additional information is needed to evaluate compliance with these items.

1. Item 4.2.1 - Periodic Maintenance Program for Reactor Trip Breakers.

1.1 Information Request for Item 4.2.1

Included with the submittal is a copy of the Beaver Valley 1 Preventative Maintenance Procedure for the Reactor Trip Switchgear effective March 9, 1983, which is in substantial conformity with the Westinghouse recommendations. However, the submittal states that "when the WOG program is finalized, we will review the program and adopt the preventative maintenance recommendations determined necessary to maintain the reactor trip breakers." Do you intend to adopt the Owners Group program in total ? If not, identify any exceptions that may be taken with respect to the Westinghouse recommendations for the maintenance of the breakers and provide appropriate justification.

1.2 Criteria for Evaluating Compliance With Item 4.2.1

The Beaver Valley 1 Reactor Trip System utilizes Westinghouse DB-50 circuit breakers. The primary criteria identified for an acceptable maintenance program for this breaker is contained in the Westinghouse Maintenance Program for DB-50 Trip Switchgear[2].

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1. Letter J. J. Carey, Duquesne Light, to D. G. Eisenhut, NRC, Response to Generic Letter 83-28 November 4, 1983.
2. Maintenance Program for DB-50 trip switchgear Revision 0, October 14, 1983.

Specifically, the criteria used to evaluate compliance should include those items in the Westinghouse program that relate to the safety function of the breaker supplemented by those measures that must be taken to accumulate data for trending.

2. Item 4.2.2 - Trending of Reactor Trip Breaker Parameters to forecast degradation of operability.

2.1 Information Request for Item 4.2.2

a. The submittal states that the breaker time response data along with other pertinent information will be used to forecast any possible degradation in the breaker operability. You should identify what other parameters such as trip force, drop-out voltage and breaker insulation resistance will be used for this forecast.* You should also provide verification that the selection of parameters is sufficient to track all of the relevant factors that give indication of degradation of the breaker safety related function and that the breaker time response measurement includes the operating time of the under-voltage trip attachment.

b. The submittal states that trend report results of the reactor trip breakers will be issued periodically to the plant's upper management staff and significant degradation found during breaker trending will be immediately identified for corrective action. Please provide a discussion of the technical criteria to be used for evaluating trend data. It should include a description of the use of acceptance limits, establishment of baseline values or other basis for identifying significant degradation of the breaker. It should also indicate the schedule or guideline for scheduling evaluation of the trend data.

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*Four parameters have been identified as trendable and are included in the criteria for evaluation. These are (1) Under-voltage trip attachment dropout voltage, (2) trip force, (3) breaker response time for under-voltage trip, and (4) breaker insulation resistance.