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

OCT 11 1985

Docket No. 50-412

Mr. John J. Carey
Vice President, Nuclear Construction
Duquesne Light Company
Robinson Plaza Building, No. 2, Suite 210
PA Route 60
Pittsburgh, Pennsylvania 15205

Dear Mr. Carey:

Subject: Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as the Structural Design Basis for Beaver Valley Power Station, Unit 2

In a letter to H. R. Denton dated February 24, 1984, Duquesne Light Company (DLC) requested a partial exemption from General Design Criteria (GDC) 4 of Appendix A to 10 CFR Part 50. Subsequently, by letters dated May 31, July 16, and November 5, 1984, in response to a letter from NRC dated April 10, 1984, DLC provided technical justification for eliminating large primary loop piping ruptures as a structural design basis. Futhermore, by a letter dated July 9, 1985, DLC requested a schedular partial exemption from GDC 4.

The staff has completed its technical review of the DLC submittal regarding the application of the "leak-before-break" technology as an alternative to providing protective devices against the dynamic effects resulting from postulated ruptures of the Beaver Valley, Unit 2 primary coolant loops.

We find that the primary coolant system piping geometry, materials, and loads are very similar to and within the envelope of those evaluated in Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops." We conclude that Duquesne Light Company has provided technical justification for not providing protective devices against the dynamic effects of postulated pipe breaks in support of its request for schedular exemption from a portion of the requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50.

Therefore, on the basis of the staff's evaluation of your submittals, the Commission has granted your schedular exemption request for Beaver Valley, Unit 2 .

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A copy of the exemption is enclosed. The exemption granted will become effective upon the date of issuance. The enclosed exemption is being forwarded to the Office of the Federal Register for publication, accordingly.

Sincerely,

ORIGINAL SIGNED BY

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure: As stated

cc: See next page

*Previously concurred on by:

DL:LB #3	LA:DL:LB #3	RAB:DSI	DE	OELD	OELD	DL:LB #3
BKSingh/dh	JLee	FCongel	RBosnak	LDewey	JSCinto	GWKnighton
*8/6 /85	*8/6 /85	*8/7/85	* 8/8 /85	*9/3 /85	9/ /85	* 9/9 /85

DE	~~DL:AD/L~~	DE
BLiaw	TMNovak	JPKnight
9/18/85	9/24/85	9/20/85

J. Scinto authorized L. Dewey to sign for OELD. However, he approved it by a memo dated 9/3/85.

Mr. J. J. Carey
Duquesne Light Company

Beaver Valley 2 Power Station

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REQUEST FOR EXEMPTION FROM A PORTION OF GENERAL DESIGN CRITERION 4 OF APPENDIX A TO 10 CFR PART 50 REGARDING THE NEED TO ANALYZE LARGE PRIMARY LOOP PIPE RUPTURES AS THE STRUCTURAL DESIGN BASIS FOR BEAVER VALLEY POWER STATION, UNIT 2

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UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
DUQUESNE LIGHT COMPANY, ET AL.	)	Docket No. 50-412
	)	
(Beaver Valley Power Station, Unit 2)	)	

EXEMPTION

I.

On October 20, 1972, the Duquesne Light Company, as an applicant and agent for the owners, tendered an application for a license to construct Beaver Valley Power Station (BVPS), Unit 2 with the Atomic Energy Commission (currently the Nuclear Regulatory Commission, or the Commission). Following a public hearing before the Atomic Safety and Licensing Board, the Commission issued Construction Permit No. CPPR-105 on May 3, 1974. The facility is a three-loop pressurized water reactor, containing a Westinghouse Electric Company nuclear steam supply system, located at a site in Beaver County, Pennsylvania.

On May 18, 1983, the applicants* tendered an application for an operating license, which is currently in the licensing review process.

II.

The Construction Permit issued for constructing the facility provides that the facility is subject to all rules, regulations and Orders of the Commission. This includes General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50. GDC 4 requires that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation,

*The applicants are Duquesne Light Company, Ohio Edison Company, The Cleveland Electric Illuminating Company and the Toledo Edison Company.

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maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit.

By letter dated July 16, 1984, the applicants for Beaver Valley, Unit 2, submitted a report (Reference 1) on the technical bases for eliminating large primary loop piping ruptures as a structural design basis. This submittal, supplemented by letters dated November 5, 1984 and July 9, 1985, was made in support of a request for a schedular exemption to General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50 in regard to the need for protection against dynamic effects from postulated pipe breaks. By means of deterministic fracture mechanics analyses, the applicants contend that postulated double-ended guillotine breaks (DEGBs) of the primary loop reactor coolant piping will not occur in the Beaver Valley, Unit 2, and therefore the dynamic loading effects associated with such breaks need not be considered as a design basis for installing protective devices such as pipe whip restraints and jet impingement shields to guard against the dynamic effects associated with such postulated breaks and need not be considered in the design of main loop piping, branch lines and branch line supports. No other changes in design requirements are addressed within the scope of the referenced reports;e.g., no changes to the definition of a LOCA nor its relationship to the regulations addressing design requirements for ECCS (10 CFR 50.46), overall containment system (GDC 16, 50), other engineered safety features and the conditions for environmental qualification of equipment (10 CFR 50.49).

III

The Commission's regulations require that applicants provide protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping. Protective measures include physical isolation from postulated pipe rupture locations if feasible or the installation of pipe whip restraints, jet impingement shields or compartments. In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from these large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that an exemption from the regulations would be acceptable as an alternative for resolution of USI A-2 for 16 facilities owned by 11 licensees in the Westinghouse Owner's Group (one of these facilities, Fort Calhoun, has a Combustion Engineering nuclear steam supply system). This NRC staff position was stated in Generic Letter 84-04, published on February 1, 1984 (Reference 2). The generic letter states that the affected licensees must justify an exemption to GDC 4 on a plant-specific basis. Other PWR applicants or licensees may request similar exemptions from the requirements of GDC 4 provided that they submit an acceptable technical basis for eliminating the need to postulate pipe breaks.

The acceptance of an exemption was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics

techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the DEGB or its equivalent. The concept underlying such analyses is referred to as "leak-before-break" (LBB). There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes or piping systems and the application of appropriate remedial measures, if indicated, can reduce the probability of catastrophic failure to insignificant values.

Advanced fracture mechanics technology was applied in topical reports (References 3, 4, and 5) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group. Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low such that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields. The staff's Topical Report Evaluation is attached as Enclosure 1 to Reference 2.

Probabilistic fracture mechanics studies conducted by the Lawrence Livermore National Laboratories (LLNL) on both Westinghouse and Combustion Engineering nuclear steam supply system main loop piping (Reference 6) confirm that both the probability of leakage (e.g., undetected flaw growth through the pipe wall by

fatigue) and the probability of a DEGB are very low. The results given in Reference 6 are that the best-estimate leak probabilities for Westinghouse nuclear steam supply system main loop piping range from 1.2×10^{-8} to 1.5×10^{-7} per plant year and the best-estimate DEGB probabilities range from 1×10^{-12} to 7×10^{-12} per plant year. Similarly, the best-estimate leak probabilities for Combustion Engineering nuclear steam supply system main loop piping range from 1×10^{-8} per plant year to 3×10^{-8} per plant year, and the best estimate DEGB probabilities range from 5×10^{-14} to 5×10^{-13} per plant year. These results do not affect core melt probabilities in any significant way.

During the past few years it has also become apparent that the requirement for installation of large, massive pipe whip restraints and jet impingement shields is not necessarily the most cost effective way to achieve the desired level of safety, as indicated in Enclosure 2, Regulatory Analysis, to Reference 2. Even for new plants, these devices tend to restrict access for future inservice inspection of piping; or if they are removed and reinstalled for inspection, there is a potential risk of damaging the piping and other safety-related components in this process. If installed in operating plants, high occupational radiation exposure (ORE) would be incurred while public risk reduction would be very low. Removal and reinstallation for inservice inspection also entail significant ORE over the life of a plant.

IV.

The primary coolant system of Beaver Valley, Unit 2, described in Reference 1, has three (3) main loops each comprising a 33.9 inch diameter hot leg, a 36.2 inch diameter crossover leg and 32.2 inch diameter cold leg piping. The material in the primary loop piping is cast stainless steel (SA 351 CF8A). In its review

of Reference 1, the staff evaluated the Westinghouse analyses with regard to:

- the location of maximum stresses in the piping, associated with the combined loads from normal operation and the SSE;
- potential cracking mechanisms;
- size of through-wall cracks that would leak a detectable amount under normal loads and pressure;
- stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load;
- margin based on crack size; and
- the fracture toughness properties of thermally-aged cast stainless steel piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in its Topical Report Evaluation, Enclosure 1 to Reference 2, Section 4.1, "NRC Evaluation Criteria," and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments and safe-ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits and controls; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

(3) A through-wall crack should be postulated at the highest stressed locations determined from (1) above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(4) It should be demonstrated that the postulated leakage crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be determined by a crack stability analysis, i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size margin should be determined by comparing the leakage-size crackto the critical-size crack. Under normal plus SSE loads, it should be demonstrated that there is adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses, and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

(6) The materials data provided should include types of materials and materials specifications used for base metal, weldments and safe-ends, the materials properties including the J-R curve used in the analyses,

and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, maximum crack growth).

V.

Based on its evaluation of the applicants' submittal and the analysis contained in Westinghouse Report WCAP-10565 (Reference 1), the staff finds that the applicants have presented an acceptable technical justification, addressing the above criteria, for not installing protective devices to deal with the dynamic effects of large pipe ruptures in the main loop primary coolant system piping of Beaver Valley, Unit 2. This finding is predicated on the fact that each of the parameters evaluated for Beaver Valley, Unit 2 is enveloped by the generic analysis performed by Westinghouse in Reference 3, and accepted by the staff in Enclosure 1 to Reference 2. Specifically:

(1) The loads associated with the highest stressed location in the main loop primary system piping are 1655 kips (axial), 12,167 in-kips (bending moment) and result in maximum stresses of about 44% of the bounding stresses used by Westinghouse in Reference 3.

(2) For Westinghouse plants, there is no history of cracking failure in reactor primary coolant system loop piping. The Westinghouse reactor coolant system primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 400 reactor-years, including five (5) plants each having 15 years of operation and 15 other plants with over 10 years of operation.

(3) The leak rate calculations performed for Beaver Valley, Unit 2, using an initial through-wall crack of 7.5 inches are identical to those of Enclosure 1 to Reference 2. The Beaver Valley plant has an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45, and it can detect leakage of one (1) gpm in one hour. The calculated leak rate through the postulated flow results in a factor of at least 10 relative to the sensitivity of the Beaver Valley, Unit 2 leak detection systems.

(4) The margin in terms of load of Beaver Valley, Unit 2 based on fracture mechanics analysis for the leakage-size crack under normal plus SSE loads is within the bounds calculated by the staff in Section 4.2.3 of Enclosure 1 to Reference 2. Based on a limit-load analysis, the load margin is at least 6 and based on the J limit discussed in (6) below, the margin is at least 3.0.

(5) The margin between the leakage-size crack and the critical-size crack was calculated by a limit load analysis. Again, the results demonstrated that a crack size margin of at least 3 exists and is within the bounds of Section 4.2.3 of Enclosure 1 to Reference 2.

(6) As an integral part of its review, the staff's evaluation of the material properties data of Reference 7 is enclosed as Appendix I to this Report. In Reference 7, data for ten (10) plants, including Beaver Valley, Unit 2, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Reference 3 report by Westinghouse. The applied J for Beaver Valley in Reference 1 was less than 3000 in-lb/in ^{2} and

hence the staff's upper bound on the applied J (refer to Appendix I, page 6) was not exceeded.

In view of the analytical results presented in Reference 1 and the staff's evaluation findings related above, the staff concludes that the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of Beaver Valley, Unit 2 is sufficiently low such that protective devices associated with postulated pipe breaks at the eight (8) locations per loop in Beaver Valley, Unit 2 primary coolant systems need not be installed. However, in order to provide the Commission with an opportunity to consider the long term aspects of the NRC staff's recent acceptance criteria of the "leak-before-break" approach, this exemption is limited to a period extending until the completion of the second refueling outage of Beaver Valley Unit 2, pending the outcome of the Commission rulemaking on the issue. By a letter dated July 9, 1985, DLC has requested such a schedular exemption.

The applicants' request does not affect the design bases for the containment, the emergency core cooling system, the environmental qualification of equipment for Beaver Valley Unit 2, or the support for heavy equipment, and does not propose to alter the design basis of reactor cavity and subcompartment pressurization from that originally performed which was based on a limited displacement DEGB. The staff agrees that this schedular exemption does not affect these matters.

The staff also reviewed the occupational radiation protection aspects of Duquesne Light Company's request for a schedular limited exemption to GDC-4 for the Beaver Valley Power Station, Unit 2. The acceptance criteria used in the evaluation were those stated in Section 12 of NUREG-0800, (SRP)

and Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." The applicants, as part of the justification for the exemption to GDC-4, have estimated an occupational dose saving for plant personnel of approximately 80 person-rem for the Unit 2 during its 40 year operating lifetime. This occupational dose estimate is based on a breakdown of occupational dose saving during inservice inspections procedures in and around the Reactor Coolant System. The staff review of the applicants' analysis shows it to be a very conservative estimate of dose saving and the staff would expect a greater dose saving due to increased efficiency in performing maintenance procedures. Therefore, from the perspective of radiation exposure and ALARA considerations, the staff finds the applicants' request acceptable.

VI.

In view of the staff's evaluation findings, conclusions, and recommendations above, the Commission has determined that, pursuant to 10 CFR 50.12(a), the following exemption is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest. The Commission hereby approves the requested schedular limited exemption from GDC 4 of Appendix A to 10 CFR Part 50, to permit the applicants to eliminate the dynamic loading effects associated with the postulated primary loop pipe breaks defined in the FSAR and as described in Part II of this report. These dynamic loading effects include pipe whip, jet impingement, and break associated dynamic transients in the main loop piping, branch lines and branch line supports. This should (1)

eliminate the need to design for pipe whip and jet impingement due to postulated primary loop pipe breaks, (2) eliminate the need for pipe whip restraints (including shims) and jet impingement shields associated with the primary loop pipe breaks defined in the Final Safety Analysis Report (FSAR), and (3) eliminate the dynamic loading effects associated with the primary loop pipe breaks defined in the FSAR on primary loop piping, branch lines and their supports. Branch line LOCA loads, including their dynamic effects, would be retained in the design basis. This exemption will expire upon completion of the GDC 4 rulemaking changes but no later than the second refueling outage.

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of the exemption will have not significant impact on the environment (50 FR 40462).

The exemption will become effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 11th day of October, 1985

REFERENCES

(1) Westinghouse Report WCAP-10565, "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for Beaver Valley, Unit 2, May 1984, Westinghouse Class 2 proprietary.

(2) NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main Loops," February 1, 1984.

(3) Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP-9558, Rev. 2, May 1981, Westinghouse Class 2 proprietary.

(4) Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation, WCAP-9787, May 1981, Westinghouse Class 2 proprietary.

(5) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981, Letter Report NS-EPR-2519, E. P. Rahe to Darrell G. Eisenhut, November 10, 1981, Westinghouse Class 2 proprietary.

(6) Lawrence Livermore National Laboratory Report, UCRL-86249, "Failure Probability of PWR Reactor Coolant Loop Piping," by T. Lo, H. H. Woo, G.S. Holman and C. K. Chou, February 1984 (Preprint of a paper intended for publication).

(7) Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," November 1983, Westinghouse Class 2 proprietary.

APPENDIX I

Evaluation of Westinghouse Report
WCAP 10456, "The Effects of Thermal Aging
on the Structural Integrity of Cast Stainless
Steel Piping for Westinghouse Nuclear Steam
Supply Systems"

INTRODUCTION

The primary coolant piping in some Westinghouse Nuclear Steam Supply Systems (NSSS) contain cast stainless steel base metal and weld metal. The base metal and weld metal are fabricated to produce a duplex structure of delta (w) ferrite in an austenitic matrix. The duplex structure produces a material that has a higher yield strength, improved weldability and greater resistance to intergranular stress corrosion cracking than a single phase austenitic material. However, as early as 1965 (Ref.1), it was recognized that long time thermal aging at primary loop water temperatures (550°F-650°F) could significantly affect the Charpy impact toughness of the duplex structured alloys. Since the Charpy impact test is a measure of a material's resistance to fracture, a loss in Charpy impact toughness could result in reduced structural stability in the piping system.

The purpose of Report WCAP 10456 is to evaluate whether cast stainless steel base metal and weld metal containing postulated cracks will be sensitive to unstable fracture during the 40 year life of a nuclear power plant. In order to determine whether a piping system will behave

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in such a fashion, the pipe materials' mechanical properties, design criteria and method of predicting failure must be established. In this evaluation, the NRC staff assesses the mechanical properties of thermally aged cast stainless steel pipe materials, which are reported in WCAP-10456.

DISCUSSION

1. Weld Metal

Report WCAP-10456 refers to test results reported in a paper by Slama, et.al. (Ref. 2) to conclude that the weld metal in primary loop piping would not be overly sensitive to aging and that the aged cast pipe base metal material would be structurally limiting. In the Slama report, eight (8) welds were evaluated. The tensile properties were only slightly affected by aging. The Charpy U-notch impact energy in the most highly sensitive weld decreased from 7daJ/cm^2 (40 ft-lbs) to near 4daJ/cm^2 (24 ft-lbs) after aging for 10,000 hours at 400°C (752°F). This change was not considered significant. The relatively small effect of aging on the weld, as compared to cast pipe material was reported to be caused by a difference in microstructure and lower levels of ferrite in the weld than in the cast pipe material.

2. Cast Stainless Steel Pipe Base Metal

Report WCAP 10456 contains mechanical property test results from a number of heats of aged cast stainless steel material and a metallurgical study, which was performed by Westinghouse, to support a statistically based model for predicting the effect of thermal aging on the Charpy impact test properties of cast stainless steel. As a result of these tests and the proposed model, Westinghouse concluded that the fracture toughness test results from one heat of material tested represents end-of-life conditions for the 10 plants surveyed. The 10 plants surveyed are identified as Plants A through J.

a. Mechanical Property Test Results Reported in WCAP-10456

Mechanical property test results on aged and unaged cast stainless steel materials were reported in papers by Landerman and Bamford (Ref. 3), Bamford, Landerman and Diaz (Ref. 4), Slama et. al; these papers were discussed in WCAP-10456. In addition, Westinghouse performed confirmatory Charpy V notch and J-integral tests on aged cast stainless steel material, which was tested and evaluated by Slama's group.

The results of these tests indicate that:

(1) The fatigue crack growth rate of aged or unaged material in air and pressurized water reactor environments were equivalent.

(2) Tensile properties were essentially unaffected except for a slight increase in tensile strength and a decrease in ductility.

(3) J-integral test results indicate that the J_{1C} and tearing modulus, T, are affected by aging.

b. Mechanism Study in WCAP 10456

The tests and literature survey conducted by Westinghouse indicate that the proposed mechanism of aging occurs in the range of operating temperatures for pressurized water reactors and the data from accelerated aging studies can be used to predict the behavior at operating temperatures.

c. Cast Stainless Steel Pipe Test

The materials data discussed in the previous section of this evaluation were obtained from small specimens. As a consequence, the J-R results are limited to relatively short crack extensions. To investigate the behavior of cast stainless steel in actual piping geometry, Westinghouse performed two experiments, one of which was with thermally aged cast stainless steel and the other test was identical except that the steel was not thermally aged.

Each pipe tested contained a throughwall circumferential crack to the extent specified in WCAP 10456. The pipe sections were closed at the ends, pressurized to nominal PWR operating pressure and then bending loads were applied.

The results of the tests were very similar, in that both pipes displayed extensive ductility, and stable crack extension. There was no observed unstable crack extension or fast fracture.

The results of the Westinghouse pipe experiments indicate that cast stainless steel, both aged and unaged, can withstand crack extensions well beyond the range of the J-R results with small specimens. However, if crack extension is predicted in an actual application of thermally aged cast stainless steel in a piping system, we believe that it is prudent to limit the applied J to 3000 in-lbs/in^2 or less unless further studies and/or experiments demonstrate that higher values are tolerable. Loss of initial toughness due to thermal aging of cast stainless steels at normal nuclear facility operating temperatures occurs slowly over the course of many years; therefore, continuing study of the aging phenomenon may lead to a relaxation of this position. Conversely, in the unlikely event that the total loss of toughness and the rate of toughness loss are greater than those projected in this evaluation, the staff will take appropriate action to limit the values to that which can be justified by experimental data. Because the aging is a slow process, the staff believes there would be sufficient time for the staff to recognize the problem and to rectify the situation. However, the staff believes this situation is highly unlikely because the staff has accepted only the lower bounds of data that were gathered among ten plants encompassing the range of materials in use.

d. Effects of Thermal Aging on Westinghouse-Supplied Centrifugally Cast Reactor Coolant Piping Reported in WCAP-10456

The reactor coolant cast stainless steel piping materials in the plants identified in WCAP-10456 as A through J, were produced to Specification SA-351, Class CF8A as outlined in ASME Code Section II, Part A, and also to Westinghouse Equipment Specification G-678864, as revised. For these materials, Westinghouse has calculated the predicted end-of-life Charpy U-notch properties, based on their proposed model. The two standard deviation end-of-life lower limit value for all the plants surveyed was greater than the Charpy U notch properties of the aged reference materials, which Westinghouse indicates represents end-of-life properties for all the plants. As a result, Westinghouse concluded that the amount of embrittlement in the aged reference material exceed the amount projected at end-of-life for all cast stainless steel pipe materials in Plants A through J.

Conclusions

On the basis of its review of the information and data contained in Westinghouse Report WCAP-10456, the staff concludes that:

1. Weld metal that is used in cast stainless steel piping system is initially less fracture resistant than the cast stainless steel base metal. However, the weld metal is less susceptible to thermal aging than the cast stainless steel base metal. Hence, at end-of-life the cast stainless steel base metal is anticipated to be the least fracture resistant material.

2. The Westinghouse proposed model may be used to predict the relative amount of embrittlement on a heat of cast stainless steel material. The two standard deviation lower confidence limit for this model will provide a useful engineering estimate of the predicted end-of-life Charpy impact properties for cast stainless steel base metal.

3. Since there is considerable scatter in J-integral test data for the heats of material tested, lower bound values for J_{1c} and T should be used as engineering estimates for the fracture resistance of the aged reference material. We believe these values should also provide a lower bound for the fracture resistance of aged and unaged weld metal. If crack extension is predicted in an actual application of cast stainless steel in a piping system, we conclude that the applied J should be limited to 3000 in-lbs/in^2 or less unless further studies and tests demonstrate that higher values are tolerable. The Westinghouse pipe tests demonstrate that this may be possible.

4. Since the predicted end-of-life Charpy impact values for the materials in Plants A through J are greater than the value measured for the aged reference material, the lower bound fracture properties for aged reference material may be used to determine the fracture resistance for the cast stainless steel material in Plants A through J.

REFERENCES

(1) F. H. Beck, E. A. Schoefer, J. W. Flowers, M. E. Fontana, "New Cast High Strength Alloy Grades by Structural Control," ASTM STP 369 (1965)

(2) G. Slama, P. Petrequin, S. H. Masson, T. R. Mager, "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Casting and Welds," presented at SMIRT 7 Post Conference Seminar 6 - Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, August 29/30, 1983, Monterey, Ca.

(3) E. I. Landerman and W. H. Bamford, "Fracture Toughness and Fatigue Characteristics of Centrifugally Cast Type 316 Stainless Steel After Simulated Thermal Service Conditions." Presented at the Winter Annual Meeting of the ASME, San Francisco, Ca., December 1978 (MPC-8 ASME)

(4) W. H. Bamford, E. I. Landerman and E. Diaz, "Thermal Aging of Cast Stainless Steel and Its Impact on Piping Integrity." Presented at ASME Pressure Vessel and Piping Conference, Portland, Oregon, June 1983.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUQUESNE LIGHT COMPANY, ET AL.

BEAVER VALLEY UNIT 2

DOCKET NO. 50-412

ENVIRONMENTAL ASSESSMENT AND

FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an Exemption from a portion of the requirements of General Design Criterion (GDC) 4 (10 CFR 50, Appendix A) to the applicants* for Beaver Valley Unit 2, located at the applicants' site in Beaver County, Pennsylvania.

ENVIRONMENTAL ASSESSMENT:

Identification of Proposed Action: The Exemption would permit the applicants not to install the pipe whip restraints and jet impingement shields and not to consider the dynamic effects associated with postulated pipe breaks in the Beaver Valley Unit 2 primary coolant system, on the basis of advance calculational methods for assuring that piping stresses would not result in rapid piping failure; i.e., pipe breaks.

*The applicants are Duquesne Light Company, Ohio Edison Company, the Cleveland Illuminating Company and the Toledo Edison Company.

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Need for Proposed Action: The proposed Exemption is needed in order for the applicants not to consider the dynamic loading effects associated with the postulated full flow circumferential and longitudinal pipe ruptures in the main loop primary coolant system. These dynamic loading effects include pipe whip, jet impingement, asymmetric pressurization transients and break associated dynamic transients in unbroken portions of the main loop and connected branch lines. Therefore, the applicants would not be required to install protective devices such as pipe whip restraints and jet impingement shields related to postulated break locations in the primary coolant loops. Analysis shows that the pipe breaks, which these devices are designed to protect against, are extremely unlikely. On the other hand, the presence of these devices increase inservice inspection time in the containment and their elimination would lessen the occupational doses to workers and facilitate inservice inspections.

GDC 4 requires that structures, systems and components important to safety shall be appropriately protected against dynamic effects including the effects of discharging fluids that may result from equipment failures, up to and including a double-ended rupture of the largest pipe in the reactor coolant system (Definition of LOCA). In recent submittals the applicants have provided information to show by advanced fracture mechanics techniques that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before flaws in the piping materials can grow to critical or unstable sizes which could lead to large break areas such as the double-ended guillotine break or its equivalent. The NRC staff has reviewed and accepted the applicants' conclusion. Therefore, the NRC staff

agrees that double-ended guillotine break in the primary pressure coolant loop piping, and its associated dynamic effects, need not be required as a design basis accident for pipe whip restraints and jet shields; i.e., the restraints and jet shields are not needed. Accordingly, the NRC staff agrees that a partial exemption from GDC 4 is appropriate.

Environmental Impact of the Proposed Action: The proposed Exemption would not affect the environmental impact of the facility. No credit is given for the restraints and shields to be eliminated in calculating accident doses to the environment. While the jet impingement barriers and pipe whip restraints would minimize the damage from jet forces and whipping from a broken pipe, the calculated limitation on stresses required to support this Exemption assures that the probability of pipe breaks which could give rise to such forces are extremely small; thus, the pipe whip restraints and jet shield would have no significant effect on the overall plant accident risk.

The Exemption does not otherwise affect radiological plant effluents. Likewise, the relief granted does not affect non-radiological plant effluents, and has no other environmental impact. The elimination of the pipe whip restraints and jet impingement shields would tend to lessen the occupational doses to workers inside containment. Therefore, the Commission concludes that there are no significant radiological or non-radiological impacts associated with the Exemption.

The proposed Exemption involves design features located entirely within the restricted area as defined in 10 CFR 20. It does not affect plant non-radioactive effluents and has no other environmental impact. Therefore, the Commission concludes that there are no non-radiological impacts associated with this proposed Exemption.

Since we have concluded that there are no measurable negative environmental impacts associated with this Exemption, any alternatives would not provide any significant additional protection of the environment. The alternative to the exemption would be to require literal compliance with GDC 4.

Alternative Use of Resources: This action does not involve the use of resources not previously considered in the Final Environmental Statement (Operating License) for Beaver Valley Unit 2.

Agencies and Persons Contacted: The NRC staff reviewed the applicants' request and applicable documents referenced therein that support this Exemption for Beaver Valley Unit 2. The NRC did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

The Commission has determined not to prepare an environmental impact statement for this action. Based upon the environmental assessment, we conclude that this action will not have a significant effect on the quality of the human environment.

For details with respect to this action, see the requests for exemption dated February 24, May 31, July 16, and November 5, 1984, and July 9, 1985. These documents, utilized in the NRC staff's technical evaluation of the exemption request, are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Local Public Document Room at the B. F. Jones Memorial Library, 663 Franklin Avenue,

Aliquippa, Pennsylvania 15001. The staff's technical evaluation of the request will be published with the exemption (if the exemption is granted) and will also be available for inspection at both locations listed above.

Dated at Bethesda, Maryland this 26th day of September, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

LB#3/DL	LB#3/DL	OELD	LB#1/DL	AD/L/DL
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