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

JUN 5 1985

Docket No.: 50-423

Mr. John F. Opeka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

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 Millstone Nuclear Power Station, Unit 3

In a letter to me dated September 12, 1984, Northeast Nuclear Energy Company (NNECo) requested an exemption from a portion of the requirements of General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50. You provided the Westinghouse report "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Millstone Unit 3," WCAP-10586 (Westinghouse Non-Proprietary) and WCAP-10587 (Westinghouse Proprietary) as an enclosure to this letter which serves as the technical basis in support of the request. The Westinghouse report addresses the "leak-before-break" concept as an alternative to providing protective devices against the dynamic effect of postulated ruptures in the primary coolant loops. Your submittal also provided a value-impact analysis associated with your exemption request.

In a letter to me dated October 18, 1984, Northeast Nuclear Energy Company submitted a Modification to Request for Exemption from General Design Criterion 4 to disregard all reference to considerations for including the P-1 snubbers in the exemption request.

Furthermore, by letter to me dated May 7, 1985, you requested that a partial exemption to GDC-4 be granted for the first two cycles of operation.

On the basis of the staff's evaluation of these submittals the Commission has granted your exemption request for Millstone Nuclear Power Station, Unit 3 for a period ending at the completion of the second refueling outage of Millstone Unit 3, pending the outcome of the Commission's ongoing rulemaking on this subject. The staff has received your request for construction permit (CP) amendment for Millstone, Unit 3 dated March 1, 1985 addressing your exemption request. 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,

L. Olshan

for B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures: As stated

cc: See next page

No legal objection 5/17/85

*SEE PREVIOUS PAGE FOR CONCURRENCES

LB#1:DL	OELD	AD/A:DL	D:DL		
BJYoungblood		TNovak	HThompson		
05/21 /85	05/ /85	05/21 /85	05/ /85		
MEB:DE	MEB:DE	CSB:DSI	AD/RS:DSI	RAB:DSI	AD:RP:DSI
*FCherny	*RBosnak	*JKudrick	RHouston	FCongel	DMuller
03/25/85	04/14/85	05/07/85	05/10 /85	05/10/85	05/10 /85
LB#1:DL	LB#1:DL	MTEB:DE	MEB:DE	AD:MCET:DE	AD:CSE:DE
*EDoolittle:kab	*MRushbrook	*RKlecker	*BDLiaw	*WJohnston	JKnight
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MILLSTONE

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UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
NORTHEAST NUCLEAR ENERGY COMPANY	)	Docket No. 50-423
	)	
(Millstone Nuclear Power Station, Unit 3)	)	

EXEMPTION

I.

On February 10, 1973, the application for a license to construct Millstone Nuclear Power Station, Unit 3 (Millstone Unit 3 or the facility) tendered by Millstone Point Company and joint applicants was docketed by 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-13 permitting the construction of Unit 3 on August 9, 1974. The facility is a pressurized water reactor, containing a Westinghouse Electric Company nuclear steam supply system, located at the licensee's site in the town of Waterford, New London County, Connecticut on the north shore of Long Island Sound.

On October 29, 1982, the licensee tendered an application for an Operating License for the facility, currenlty in the licensing review process.

II.

The Construction Permit issued for constructing the facility provides, in pertinent part, 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, 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 a submittal dated September 12, 1984, the applicant enclosed Westinghouse Report WCAP-10586 (Wesinghouse Non-Proprietary) and WCAP-10587 (Westinghouse Proprietary) (Reference 1) containing the technical basis for their request to: (1) eliminate the need to design for pipe whip, jet impingement and the asymmetric effects of cavity pressurization due to primary loop pipe breaks; (2) eliminate the need for pipe whip restraints and jet impingement shields on primary loop piping; (3) eliminate primary loop LOCA load evaluation on primary loop piping, branch line piping and branch line supports (branch line LOCA loads would be retained in the design basis); (4) eliminate the need to include primary loop LOCA loads in the design of the reactor coolant pump P1 snubber in loops 1 and 2 (2 out of 44 snubbers).

By a subsequent submittal dated October 18, 1984, the applicant submitted a modification to request for exemption from General Design Criterion 4 to disregard all reference to considerations for including the P-1 snubbers in the exemption request.

The applicant also stated in their submittal that the exemption request does not affect the emergency core cooling system design bases, containment and subcompartment design bases, equipment qualification bases and engineered safety features systems response. The applicant also stated that the design of the reactor coolant system heavy component supports will continue to assume a double ended primary loop pipe break with a break area equal to that which would occur if pipe whip restraints were installed.

The applicant provided a value-impact analysis in its September 12th submittal which, together with the technical information contained in the Reference 1 report, provided a comprehensive justification for requesting a partial exemption from the requirements of GDC 4.

Finally, by letter from J. F. Opeka to B. J. Youngblood dated May 7, 1985, the applicant requested that a partial exemption to GDC 4 be granted for the first two cycles of operation.

From the deterministic fracture mechanics analysis contained in the technical information furnished, the applicant concluded that the dynamic loading effects associated with postulated double-ended guillotine breaks (DEGB) and longitudinal breaks in the primary loop coolant piping in Millstone Unit 3, need not be considered as a design basis. These dynamic loading effects include pipe whip, jet impingement, asymmetric pressurization transients and break associated transients.

Therefore, structures such as pipe whip restraints and jet impingement shields, to guard against the dynamic effects associated with such postulated breaks may be eliminated.

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 barriers. 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 of piping systems and the application of appropriate remedial measures, if indicated, can reduce the probability of catastrophic failure to very small 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 Millstone 3 described in Reference 1, has four (4) 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 should be determined by comparing leakage-size crack to critical-size cracks. 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 analysis contained in Westinghouse Report WCAP-10587 (Reference 1), the staff finds that the applicant has presented an acceptable technical justification for eliminating the dynamic loading effects associated with the postulated full flow area circumferential and longitudinal pipe ruptures in the main loop primary coolant system of Millstone 3.

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 (branch line LOCA loads would be retained in the design basis). This finding is predicated on the fact that each of the parameters evaluated for Millstone 3 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 2032 kips (axial), 28,789 in-kips (bending moment) and result in maximum stresses of about 78% 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 Millstone 3, using an initial through-wall crack of 7.5 inches are identical to those of Enclosure 1 to Reference 2. The Millstone 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 flaw results in a factor of at least 10 relative to the sensitivity of the Millstone 3 detection systems.

(4) The margin in terms of load of the Millstone unit based on fracture mechanics analyses 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 about 2.8 and based on the J limit discussed in (6) below, the margin is at least 1.5.

(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 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 Safety Evaluation Report. In Reference 7, data for ten (10) plants, including the Millstone unit, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Reference 1 report by Westinghouse. The applied J for Millstone 3 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 Millstone 3 is sufficiently low such that protective devices associated with postulated pipe breaks at the eight (8) locations per loop in the Millstone 3 primary coolant system 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 completion of the second refueling outage of Millstone Unit 3 pending the outcome of Commission rulemaking on this issue.

Eliminating the need to consider these dynamic loads for this particular application has not affected the design bases for the emergency core cooling system, the overall containment, the response of engineered safety features systems, or the environmental qualification of equipment for Millstone 3. Also, it does not propose to alter the design bases of reactor cavity and subcompartment pressurization from that originally approved, which were based on the governing piping ruptures.

The staff also reviewed the value-impact analysis provided by the applicant in their September 12, 1984 submittal for not providing protective structures against the dynamic loading effects of postulated reactor coolant system loop pipe breaks to assure as low as reasonably achievable (ALARA) exposure to plant personnel. Consideration was given to design features for reducing doses to personnel who must operate, service and maintain the Millstone 3 instrumentation, controls, equipment, etc. The Millstone Unit 3 value-impact analysis show that the elimination of protective devices for RCS pipe breaks will save an occupational dose for plant personnel of approximately 200 person-rem for the facility over its operating lifetime. The staff review of the analysis shows it to be a reasonable estimate of dose savings. Therefore, with respect to occupational exposure, the staff finds that there is a radiological benefit to be gained by eliminating the need for the protective structures.

IV.

In view of the staff's evaluation findings, conclusions, and recommendations above, the Commission has determined that, pursuant to 10 CFR 50.12(a), this 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 limited schedular exemption from GDC 4 of Appendix A to 10 CFR Part 50, to eliminate the requirement to install protective devices and the requirement to consider dynamic effects and loading conditions, as detailed in Part II of this exemption, associated with postulated pipe breaks of the eight (8) locations per loop in the Millstone Unit 3 primary coolant system. This exemption is effective for a period ending at the completion of the second refueling outage, pending the outcome of rulemaking on this subject.

Pursuant to 10 CFR 51.31, the Commission has determined that the issuance of the exemption will have no significant impact on the environment (50 FR 21954).

The exemption will become effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Dated at Bethesda, Maryland
this 5th day of June 1985.

*SEE PREVIOUS PAGE FOR CONCURRENCES

LB#1:DL	OELD	AD:L:DL	D:DL
BJYoungblood	*JScinto	TNovak	HThompson
05/ /85	05/ /85	05/21/85	06/4 /85

MEB:DE	MEB:DE	CSB:DSI	AD:RS:DSI	RAB:DSI	AD:RP:DSI
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LB#1:DL	LB#1:DL	MTEB:DE	MEB:DE	AD:MCET:DE	AD:CSE:DE
*EDoolittle:kab	*MRushbrook	*RKlecker	*BDLiaw	*WJohnston	*JKnight
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(1) Westinghouse Report WCAP-10587, "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for Millstone Unit 3, June 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 Reponse 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.

NOTE: Non-proprietary versions of References 1, 3, 4, 5 and 7 are available in the NRC Public Document Room as follows:

(1) WCAP-10586, non-proprietary
(2) WCAP-9570
(3) WCAP-9788
(5) Non-proprietary version attached to the Letter Report
(6) WCAP-10457

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 in such a fashion, the pipe materials' mechanical properties, design criteria and method of predicting failure must be established. In this evaluation, we will assess the mechanical properties of thermally aged cast stainless steel pipe materials, which are reported in Report 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 $7 daJ/cm^2$ (40 ft-lbs) to near $4 daJ/cm^2$ (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 concludes that the fracture toughness test results from one heat of material tested represents end-of-life conditions for the ten (10) plants surveyed. The ten (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 which were reported in a paper by Landerman and Bamford (Ref. 3), Bamford, Landerman and Diaz (Ref. 4), Slama et. al. (Ref. 2) were discussed in Report 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 et. al.

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. <u>Mechanism Study in WCAP 10456</u>

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 the 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 V-notch properties, based on their proposed model. The two (2) 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

Based on our review of the information and data contained in Westinghouse Report WCAP 10456, we conclude 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.

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