PDR

DEC 2 2 1986

Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Midland Square, 4th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has completed the review of Northern States Power Company's (the licensee) request for an exemption to allow the application of the "leak-before-break" technology as a basis for the elimination of protective devices (i.e., pipe whip restraints, jet impingement barriers, and other related changes) of the primary reactor coolant systems at the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2. These protective devices were installed to mitigate the dynamic effects resulting from postulated large pipe ruptures. The technical information was provided by the licensee's letters dated October 24, 1984, October 21, and November 5, 1985 and supplemented by letter dated September 10, 1986 in response to staff concerns.

On April 11, 1986 a final rule was published in the Federal Register (51 FR 12502) amending 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 4 that became effective on May 12, 1986. The revision of GDC 4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant system. The staff has completed the review of the licensee's submittals and concludes that the analysis of piping of primary coolant systems at the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 is adequate and demonstrates compliance with the GDC 4 as amended. Therefore, an exemption to GDC 4 of Appendix A of 10 CFR Part 50 as amended that was requested by the licensee prior to the effective date of the rule is not necessary. On this basis, the removal of pipe whip restraints, jet impingement barriers, and other associated plant hardware may be implemented at your convenience. Our safety evaluation addressing this matter is enclosed.

This action completes our work effort under TAC Nos. 08731 and 08732.

Sincerely,

/s/

Dominic C. DiIanni, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosure:
Safety Evaluation

cc's: See Next Page

Office:	LA/PAD#1	PM/PAD#1 DCD	PD/PAD#1 DCD/for
Surname:	PShuttleworth	DDiIanni/tg	GLear
Date:	12/ /86	12/22/86	12/22/86

Mr. D. M. Musolf
Northern States Power Company

Prairie Island Nuclear Generating
Plant

cc:
Gerald Charnoff, Esq.
Shaw, Pittman, Potts and Trowbridge
2300 N. Street, N.W.
Washington, DC 20037

Dr. J. W. Ferman
Minnesota Pollution Control Agency
520 LaFayette Road
St. Paul, MN 55155

Mr. E. L. Watzl, Plant Manager
Prairie Island Nuclear Generating Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Jocelyn F. Olson, Esq.
Special Assistant Attorney General
Minnesota Pollution Control Agency
1935 W. County Road, B2
Roseville, Minnesota 55113

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
1719 Wakonade Drive East
Welch, Minnesota 55089

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Office of Executive Director for
Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. William Miller, Auditor
Goodhue County Courthouse
Red Wing, Minnesota 55066

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE ELIMINATION OF LARGE PRIMARY LOOP RUPTURES AS A DESIGN BASIS

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

DOCKET NOS. 50-282 & 50-306

BACKGROUND

By letters dated October 24, 1984, October 21, and November 5, 1985, Northern States Power Company (the licensee) provided technical information and requested an exemption to allow the application of "leak-before-break" technology as a basis for the elimination of protective devices installed to mitigate the dynamic effects resulting from postulated ruptures of Prairie Island Units 1 and 2 primary coolant loops. The licensee submitted Westinghouse reports WCAP-10639, WCAP-10929, and WCAP-10931 as technical bases for the request. By letter dated September 10, 1986, the licensee submitted Revision 1 of Westinghouse reports WCAP-10929 and WCAP-10931 in response to staff concerns on Prairie Island Unit 2. The submittals were made in support of a request for an exemption to General Design Criterion (GDC) 4 in regard to the need for protection against dynamic effects from postulated primary loop pipe breaks.

On April 11, 1986, a final rule was published (51 FR 12502), effective May 12, 1986, amending 10 CFR Part 50, Appendix A, GDC 4. The revised GDC 4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures of primary coolant loop piping in pressurized water reactors. In the "summary" section of the final rule, it is stated that the new technology reflects an engineering advance which allows simultaneously an increase in safety, reduced worker radiation exposures and lower construction and maintenance costs. Implementation permits the removal of pipe whip restraints and jet impingement barriers as well as other related changes in operating plants, plants under construction and future plant designs. Containment design, emergency core cooling and environmental qualification requirements are not influenced by this modification. In the "supplementary information" section of the final rule, it is stated that acceptable technical procedures and criteria are defined in NUREG-1061, Volume 3, dated November 1984 and entitled "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks."

With the revised GDC 4, the exemption originally requested is no longer necessary. Using the criteria in NUREG-1061, Volume 3, the staff has reviewed and evaluated the licensee's submittals and this report provides the staff's findings.

PRAIRIE ISLAND PRIMARY COOLANT SYSTEMS

The primary coolant systems of Prairie Island Units 1 and 2 have two main loops each comprising a 34.6 inch diameter (outside) hot leg, a 36.9 inch diameter crossover leg and 32.8 inch diameter cold-leg piping. The materials for the primary loop piping are wrought stainless steel (376-TP316) and cast stainless steel (SA351-CF8M) for Prairie Island Units 1 and 2, respectively. The material for the primary loop fittings is cast stainless steel (SA351-CF8M) for Prairie Island Units 1 and 2.

STAFF EVALUATION CRITERIA

The staff's criteria for evaluation of compliance with the revised GDC 4 are provided in Chapter 5.0 of NUREG-1061, Volume 3, and are as follow:

(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 are 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 at least a factor of ten 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 at least 1.4 and 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 the leakage-size crack to the critical-size crack. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 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).

The margins cited in the staff criteria are guidelines. Their applicability is dependent upon the conservatism of the analyses performed.

STAFF EVALUATION AND CONCLUSIONS

Based on its evaluation of the analyses contained in the licensee's submittals, the staff finds that the licensee has presented an acceptable technical justification, addressing the preceding criteria, to eliminate as a design basis, the dynamic effects of large ruptures in the main loop primary coolant piping of Prairie Island Units 1 and 2. Specifically:

(1) For Prairie Island Unit 1, the loads associated with the highest stressed location in the main loop primary system piping are 2,235 kips (axial), 28,422 in-kips (bending moment) and result in maximum stresses of about 60% of the Service Level D limits specified in Section III of the ASME Code. For Prairie Island Unit 2, the loads associated with the highest

stressed location in the main loop primary system piping are 1,623 kips (axial), 28,422 in-kips (bending moment) and result in maximum stresses of about 50% of the Service Level D limits specified in Section III of the ASME Code.

(2) For the Westinghouse facilities, there is no history of cracking in reactor primary coolant system main loop piping. The reactor coolant system primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking 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 plants each having 15 years of operation and 15 other plants each with over 10 years of operation.

(3) The leak rate calculations performed for Prairie Island Units 1 and 2 used initial through-wall flaws of 7.5 inches and 7.0 inches, respectively and are within the guidelines of NUREG-1061, Volume 3. Prairie Island Units 1 and 2 have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 such that leakage of one gpm in one hour can be detected. The calculated leak rates through the postulated flaws are large relative to the staff's required sensitivity of the plant leak detection system; the margin is at least a factor of ten on leakage for Prairie Island Units 1 and 2.

(4) The margin in terms of load based on fracture mechanics analyses for the leakage-size crack under normal plus SSE loads (Service Level D loads) meets the intent of NUREG-1061, Volume 3, guidance on margins. Based on a limit-load analysis, the load margin is at least 3 for Prairie Island

Units 1 and 2. Similarly, based on the J limit, the margins are about 2 and 1.3 for Prairie Island Units 1 and 2, respectively. Although the margin on the J limit for Prairie Island Unit 2 is less than 1.4 as recommended in NUREG-1061, Volume 3, the staff has determined that if a leakage-size crack slightly less than 7.0 inch were assumed, the analysis will meet the 1.4 margin on the J limit, as well as other margins. Thus, the results demonstrated that the margin in terms of load is within the guidelines of NUREG-1061, Volume 3.

(5) The margin between the leakage-size crack and the critical-size crack was calculated by a limit load analysis. The results demonstrated that a margin of about 5 exists for Prairie Island Units 1 and 2 and is within the guidelines of NUREG-1061, Volume 3.

(6) Prairie Island Units 1 and 2 have cast stainless steel piping (and/or fittings) and associated welds in the primary coolant systems. The thermal aging properties of the Prairie Island Units 1 and 2 cast stainless steel materials are described in WCAP-10456 and WCAP-10931 (Revision 1), respectively. As an integral part of its review, the staff's evaluations of the material properties data of WCAP-10456 and WCAP-10931 (Revision 1) are enclosed as Appendices I and II, respectively, to this safety evaluation report. The applied J for Prairie Island Unit 1 in WCAP-10639 for cast stainless steel fittings and associated welds was less than 3,000 in-lb/in^2 and hence the staff's upper bound on the applied J (refer to Appendix I) was not exceeded. The applied J for Prairie Island Unit 2 in WCAP-10929 (Revision 1) for cast stainless steel piping, fittings, and associated welds was less than the maximum allowable J estimated for the specific location and hence the staff's upper bound on the applied J (refer to Appendix II) was not exceeded.

In view of the analytical results presented in the licensee's submittals, the staff concludes that the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of Prairie Island Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities need not be a design basis. Furthermore, the staff concludes that the licensee is in compliance with GDC 4, as revised.

Principal Contributors:

B. Elliot
S. Lee

Date: DEC 2 2 1986

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) contains cast stainless steel base metal and weld metal. The base metal and weld metal are fabricated to produce a duplex structure of delta (δ) 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 Westinghouse 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, 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 Material

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 statically 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), and Slama et. al (Ref. 2). 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.

(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 section 2 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 test was performed on thermally aged cast stainless steel and the other test was performed on cast stainless steel that was not thermally aged.

Each pipe tested contained a through-wall 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 had bending loads 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, the staff finds 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 from 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 finds there would be sufficient time for the staff to recognize the problem and to rectify the situation. However, the staff finds this to be a highly unlikely situations because the staff has accepted only the lower bounds of data that were gathered among 10 plants encompassing the range of materials in use.

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

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. The staff believes 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, the staff concludes 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," ASME Pressure Vessel and Piping Conference, Portland, Oregon, June 1983, American Society of Mechanical Engineers, N.Y., Circumferential Cracks in Pressure and Piping, Vol.II, 1984, PP. 137-172.

Appendix II

Evaluation of Westinghouse Report
WCAP 10931, Revision 1, "Toughness Criteria
For Thermally Aged Cast
Stainless Steel"

Introduction

Westinghouse Report WCAP 10931, Revision 1, "Toughness Criteria For Thermally Aged Cast Stainless Steel," provides criteria for evaluating the fracture resistance of thermally aged cast stainless steel piping for Westinghouse Nuclear Steam Supply Systems. The criteria in the report are divided into three categories. Based on the predicted end-of-life KCU impact energy value for the heat of material, the material is considered to be either category 1, 2 or 3. The category 1 fracture toughness properties and the equations for predicting end-of-life KCU impact energy were previously documented in a Westinghouse Report (Ref. 1). The staff's review of this report is contained in Appendix I.

As discussed in WCAP 10931 and reference 1, cast stainless steel is a two phase alloy consisting of austenite and ferrite. It has been found that the chrome enriched ferrite of the two phase alloy becomes hardened and embrittled when thermally aged at primary loop water temperatures (550-600°F).

Discussion

WCAP 10931, Revision 1, describes the fracture properties of materials which have fracture resistance below that of the reference material discussed in Ref. 1. These materials are either category 2 or 3. In addition, WCAP 10931, Revision 1, provides estimates and the bases for the uncertainty in the end-of-life KCU impact energy prediction equations and the uncertainty in the fracture properties.

The Category 2 and 3 fracture toughness properties are based on the extrapolation of existing aged reference material for higher ferrite content and a fracture energy model described in Ref. (2). The aged reference material is the heat of material which was the basis for the bounding fracture properties (J-integral) reported in Ref. (1). However, Westinghouse has performed additional J-integral and Charpy impact tests on the aged reference material. The test temperatures for the J-integral tests were -320°F, -200°F, room temperature, 150°F, 200°F, 550°F and 600°F. The test temperatures for the Charpy impact tests were room temperature, 200°F and 550°F. At -320°F test temperature, the ferrite phase in cast stainless steel is completely embrittled and the J-integral test results would represent the toughness of the austenitic phase in the reference material. Hence, the J-integral data for the reference material at -320°F would conservatively represent the fully aged fracture toughness for this material. Although Charpy impact tests were not performed at -320°F, the predicted fully aged value was estimated from the J integral tests using correlations in references 3 and 4. The J-integral test at 600°F was used to establish the fracture toughness for category 1 materials.

The J-integral test at 200°F indicate that the tearing modules (Tm) at 200°F could be less than the value at 600°F. The material's Tm describes its resistance to ductile fracture after a postulated crack has initiated growth. In order for crack growth to initiate, the loads on a pipe must exceed the J_{1c} value for the material. The reduction in loads during transients in PWR's occurring at 200°F as compared to transients occurring at 600°F has been evaluated. This evaluation indicates that piping loads at 200°F are below those required to initiate crack growth. Hence, it is likely that the loads at 200°F will not be large enough to exceed the material's J_{1c} and the reduction in Tm from 600°F to 200°F is not considered significant.

To extrapolate the fully aged fracture properties from the reference material to other materials, the authors utilized a fracture energy model designated as the tortuous beam model. This model is based on the fracture energy theory described in Ref. (2) which indicates that the total fracture energy is proportional to the volume of the plastically deformed material. Test data from Charpy V-notch impact and dynamic tear tests on ASTM A 533, Grade B, Class 1 material are reported in Ref. (2) which support this theory. The tortuous beam model considers the fracture path for the cast stainless steel as a series of cracks through ductile austenite and embrittled ferrite. The model proposes that the ratio of the total fracture energy of the reference material to that of another heat of material is dependent upon the volumetric fraction of austenite and ferrite in the two materials. Using the proposed model the authors define the limiting fracture properties (J-integral and Charpy impact) for fully aged cast stainless steel. These properties have been verified by evaluating the fracture toughness data from a heat of aged cast stainless steel supplied by a Westinghouse Licensee. The Charpy impact data on this experimental heat indicates that the cast stainless steel has been fully aged. The J-integral test data from this experimental heat are higher than the values predicted by the Westinghouse model. Hence, the model appears to provide conservative estimates of the J integral properties for fully aged material.

Westinghouse has used equations proposed in Ref. (1) to predict the end-of-life KCU impact energy values for any heat of cast stainless steel. There are two equations for predicting the end-of-life KCU impact energy value of aged cast stainless steel. One equation is for CF8M material and the other is for CF8 and CF8A materials. The authors of Westinghouse Report WCAP 10931 have refined the the equations to determine the 95% lower confidence level for the aged material. The 95% confidence level for the calculated KCU value were determined by a procedure given in Reference 5. These confidence limits have been incorporated into the predicted end-of-life KCU impact energy equations to ensure that the predicted value for the aged cast stainless steel at the end-of-life of the nuclear plant is greater than that of the reference material.

The uncertainty in fracture properties for aged cast stainless steel material was determined by evaluating the uncertainty in the ferrite content. The coefficient of variation of the ferrite was determined using procedures in Reference (6) and data in Reference (7). The coefficient of variation of the ferrite was used to establish the 95% lower confidence limit for category 2 and 3 materials, which are defined below.

The category 1 material is defined as material in which the calculated KCU impact energy exceeds the upper 95% confidence KCU value for the aged reference material. The fracture properties for category 1 materials are those of the aged reference material reported in Reference 1.

The category 2 materials is defined as material in which the calculated KCU impact energy is less than the upper 95% confidence KCU value for the aged reference material and greater than the fully aged predicted mean KCU value for the reference material. The fracture properties for the category 2 material are predicted by a linear interpolation between the fracture properties of the aged reference material and the fully aged 95% lower confidence limit reference material.

The category 3 material is defined as material in which the calculated KCU impact energy is less than the fully aged predicted mean value for the reference material. The fracture properties calculated for the category 3 materials are those of the fully aged reference material at the 95% lower confidence limit.

Conclusions

The equations and methodology that are documented in WCAP 10931 Revision 1 may be utilized for establishing the fracture criteria for thermally aged cast stainless piping applicable for the leak-before-break analyses. The end-of-life KCU impact energy for CF8M cast stainless may be calculated using equation (2-17) in WCAP 10931 Revision 1. The end-of-life KCU impact energy for CF8 and CF8A material may be calculated using equation 5-2 in WCAP 10456 (Ref. 1). Based on the calculated KCU impact energy the fracture toughness for a heat of thermally aged cast stainless steel may be calculated using the procedure in Table 2-8 of WCAP 10931 Revision 1, except that for category 1 material, the maximum J applied should be limited to 3000 in-lbs/in.^{2}. The maximum J applied limit was established by the staff in its evaluation of Reference 1, which is contained in Appendix I.

References

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

(2) F.J. Witt and R.G. Berggren, "Size Effects and Energy Disposition in Impact - speciman Testing of ASTM A 533 Grade B Steel, Experimental Mechanics, Volume 11, No. 5, pp 193-201 (May 1971).

(3) F.J. Witt, "Relationships Between Charpy Impact Energies and Upper Shelf K_{IC} Values for Reactor Pressure Vessel Steels," International JOurnal of Pressure Vessels and Piping, Volume 11, pp. 47-63 (1983).

(4) S.T. Rolfe and J.M. Barsom, Fracture and Fatigue Control in Structures-Applications to Fracture Mechanics, Prentice-Hull, Chapter 6 (1977).

(5) N.R. Draper and H. Smith, Applied Regression Analysis, John Wiley and Sons, Inc., New York, 1967, pp. 21-24.

(6) E.B. Haugen, Probabilistic Approaches to Design, John Wiley and Sons, New York, 1968, pp 105-128.

(7) L.S. Aubey, et al. "Ferrite Measurement and Control in Cast Duplex Stainless Steel," Stainless Steel Castings, ASTM STP 756, American Society for Testing and Materials, 1982, pp 126-164.