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A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casing of R. E. Ginna Nuclear Power Plant for the License Renewal Program

Westinghouse

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A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R. E. Ginna Nuclear Power Plant for the License Renewal Program

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Executive Summary

Periodic volumetric inspections of the welds of the primary loop pump casings of commercial nuclear power plants are required by Section XI of the ASME Boiler and Pressure Vessel Code. These inspections are quite costly in terms of both dollars and radiation exposure (rem). To perform a volumetric inspection, complete disassembly of the pump is required. A lowering of the primary coolant water level most likely would be necessary which would, in turn, necessitate a complete core unload. Even then the volumetric inspection is very difficult. The pump casings are inspected twice prior to placing in service. When fabricated, the castings are radiographed and liquid penetrant tested. After assembly, the welds are again radiographed and liquid penetrant tested. This in-shop examination is required per Section III of the American Society of Mechanical Engineer's (ASME) Boiler and Pressure Vessel Code. The pre-service inspection criteria are the same as the in-service inspection criteria. Since no significant mechanisms exist for crack initiation and propagation, these criteria requiring that all welded surfaces are volumetrically and surface examined may not be warranted. In recognition of these facts, the ASME Code body approved Code Case N-481 that provides an alternative to the volumetric inspection requirement

The ASME Code Case N-481 (Alternate Examination Requirements for Cast Austenitic Pump Casings), allows the replacement of volumetric examinations of primary loop pump casing welds with fracture mechanics based integrity evaluations supplemented by specific visual inspections. The Westinghouse Owners Group (WOG) sponsored analyses required by the ASME Code Case N-481 is documented in WCAP-13045 and compliance to ASME Code Case N-481 was demonstrated on a generic basis.

A plant specific analysis for R. E. Ginna Nuclear Power Plant primary loop pump casings for the 40-year plant life was performed by Westinghouse and was documented in WCAP-13606 Revision 1. The objective of this report is to show compliance of the R. E. Ginna Nuclear Power Plant primary loop pump casings to ASME Code Case N-481 for the 60-year plant life (as a part of the License Renewal Program).

The evaluation documented in this report satisfies Items (d) (1), (d) (2), (d) (3), (d) (4), (d) (5), and (d) (6) of the ASME Code Case N-481. The effect of thermal aging has been evaluated and no other mechanism is known to degrade the properties of the pump casings during service and therefore Item (d) (7) of the ASME Code Case N-481, is also satisfied. Additionally, assessment of the fatigue crack growth (FCG) was performed which shows that FCG for the pump casings is not a concern for R. E. Ginna Nuclear Power Plant primary loop pump casings for the 60-year plant life.

It is concluded that the primary loop pump casings of R. E. Ginna Nuclear Power Plant are in compliance to ASME Code Case N-481 for 40-year and 60-year (as a part of the License Renewal program) plant life.

1 BACKGROUND AND OBJECTIVE

1.1 BACKGROUND

Periodic volumetric inspections of the welds of the primary loop pump casings of commercial nuclear power plants are required by Section XI of the ASME Boiler and Pressure Vessel Code (see Table IWB-2500-1, Examination Categories). These inspections are quite costly in terms of both dollars and radiation exposure (rem). To perform a volumetric inspection, complete disassembly of the pump is required. A lowering of the primary coolant water level most likely would be necessary which would, in turn, necessitate a complete core unload. Even then the volumetric inspection is very difficult. The pump casings are inspected twice prior to placing in service. When fabricated, the castings are radiographed and liquid penetrant tested. After assembly, the welds are again radiographed and liquid penetrant tested. This in-shop examination is required per Section III of the American Society of Mechanical Engineer's (ASME) Boiler and Pressure Vessel Code. The pre-service inspection criteria are the same as the in-service inspection criteria. Since no significant mechanisms exist for crack initiation and propagation, these criteria requiring that all welded surfaces are volumetrically and surface examined may not be warranted. In recognition of these facts, the ASME Code body approved Code Case N-481 that provides an alternative to the volumetric inspection requirement (Reference 2*).

The ASME Code Case, N-481 (Alternate Examination Requirements for Cast Austenitic Pump Casings), allows the replacement of volumetric examinations of primary loop pump casing welds with fracture mechanics based integrity evaluations (Item (d) of the code case) supplemented by specific visual inspections. It also requires that a report of the evaluation be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review (Item (e) of the code case). A copy of the code case is given in Appendix A.

Following approval of Code Case N-481 by the ASME, the Westinghouse Owners Group sponsored the analyses required by the code case which are applicable to the various primary loop pump casing models found in Westinghouse design nuclear steam supply systems. This work is documented in WCAP-13045 (Reference 3). Specifically, stress analyses for loadings on the pump casings were performed to support the fracture mechanics analyses for postulated flaws. Compliance to Item (d) of ASME Code Case N-481 was demonstrated on a generic basis.

However, a plant specific evaluation to demonstrate safety and serviceability is required by Code Case N-481. Since there is a variety of pumps casing models, loads and materials as discussed in WCAP-13045, it was not feasible to qualify each plant of Westinghouse design specifically to the requirements of the code case. Rather, enveloping or bounding criteria were set up whereby a specific utility, in most cases, needs only to show that the primary loop pump casings fall under the umbrella established by the analyses. The U.S. Nuclear Regulatory Commission (U.S. NRC) has approved ASME Code Case N-481 in Revision 9 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI Division I," dated April 1992.

* See Section 8.0 for a listing of references.

1.2 OBJECTIVE

It is the objective of this report to qualify the primary loop pump casings of the R. E. Ginna Nuclear Power Plant to Item (d) of ASME Code Case N-481 (Reference 2). When this report is supplemented by the visual inspections specified in the code case (Items a, b & c) and approved by the Regulatory authority, compliance to the code case will be accomplished.

The existing analysis report (Reference 4) was issued in April 1993 for the 40-year life of the plant. The objective of this report is to validate the integrity of the R. E. Ginna Nuclear Power Plant primary loop pump casings to ASME Code Case N-481 for the 60-year plant life (as a part of the License Renewal Program).

2 DESCRIPTION OF THE PRIMARY LOOP PUMP CASINGS OF R. E. GINNA NUCLEAR POWER PLANT

The primary loop pump casings of R. E. Ginna Nuclear Power Plant are Westinghouse Model 93 design. The R. E. Ginna Nuclear Power Plant pump casings are primarily fabricated from SA351 CF8M cast stainless steel with one exception. The flange outer quarter of one pump casing is fabricated from SA351 CF8 cast stainless steel. Sketches of a typical pump casing of this type along with the weld locations are shown in Figure 2-1. These sketches also contain typical dimensions.

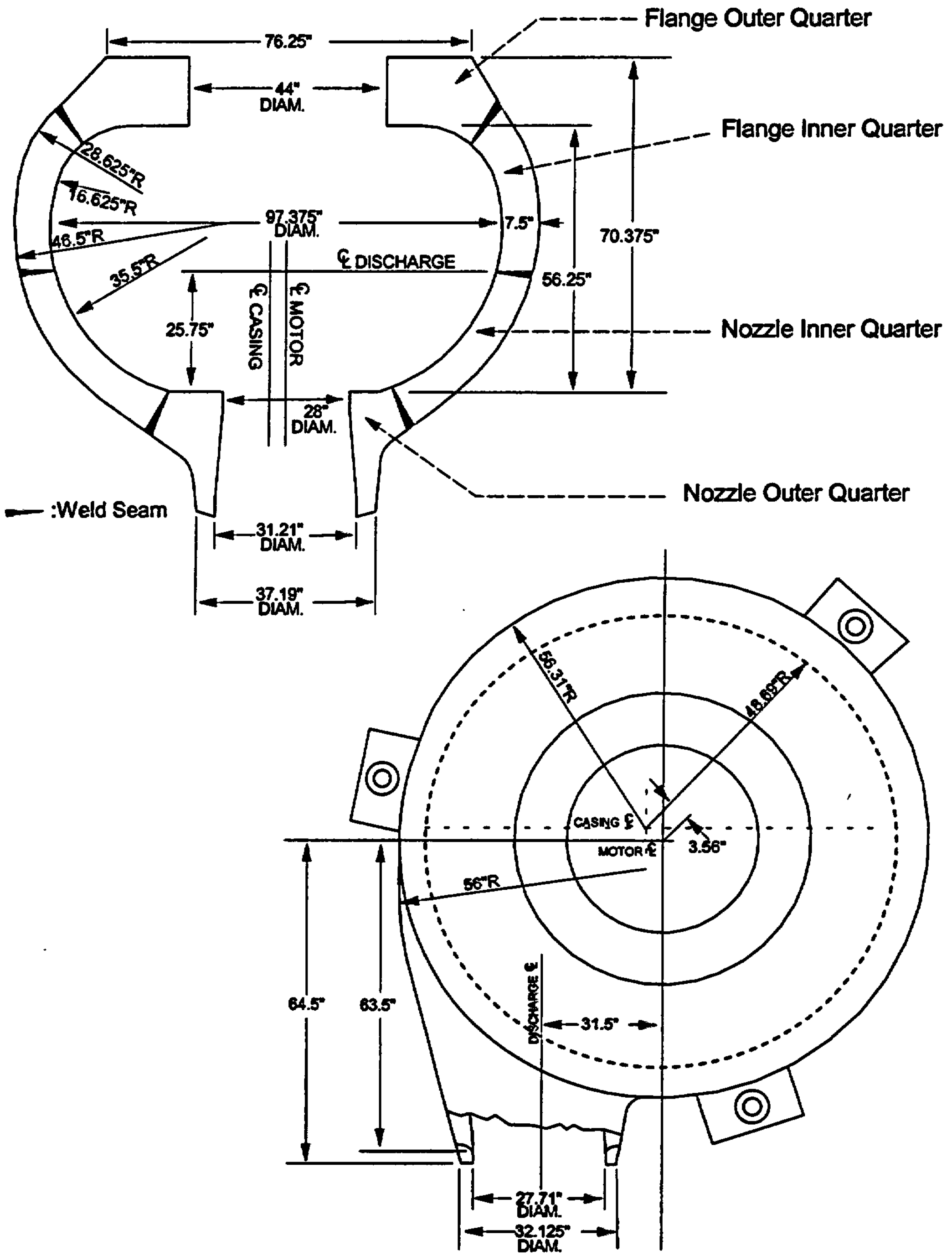

Figure 2-1 Dimensional Sketch of a Typical Model 93 Pump Casing with the Weld Seams Identified

3 FRACTURE TOUGHNESS CRITERIA FOR THE PUMP CASINGS

3.1 FRACTURE TOUGHNESS PROPERTIES AND CRITERIA FOR THE PUMP CASINGS

The R. E. Ginna Nuclear Power Plant pump casings are fabricated from SA351 CF8M except for the flange outer quarter of one pump casing which is fabricated from SA351 CF8. However, the fracture toughness criteria for SA351 CF8M is more limiting and therefore the evaluations that follow are made assuming the pump casing material is all SA351 CF8M. Values for the chemistry of each heat of material used in fabricating the R. E. Ginna Nuclear Power Plant primary loop pump casings are taken from the information shown Appendix A of WCAP-13045 (Reference 3). Predictions for fracture toughness values are based on the material chemistry content.

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In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of cast stainless steel materials. The ANL research program measured mechanical properties of cast stainless steel materials after they have been heated in controlled ovens for long periods of time. ANL compiled a data base, both from data within ANL and from international sources, of about 85 compositions of cast stainless steel exposed to a temperature range of 290-400°C (550-750°F) for up to 58,000 hours (6.5 years). From this database, ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (References 5 and 6).

ANL developed the fracture toughness estimation procedures by correlating data in the data base conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several cast stainless steel plant components removed from actual plant service. The ANL procedures produced conservative estimates that were about 30 to 50 percent less than actual measured values. The procedure developed by ANL in Reference 6 was used to calculate the fracture toughness values for this analysis. ANL research program was sponsored and the procedure was accepted (Reference 7) by the NRC.

The chemical compositions are available from CMTRs and are provided in Table 3-1. Cr_{eq}, Ni_{eq} and (δ_c) delta ferrite are obtained from Table A-2 of Reference 3 for R. E. Ginna Nuclear Power Plant primary loop pump casings.

Cr_{eq}= chromium equivalent,

Ni_{eq} = nickel equivalent,

δ_c= delta ferrite
where the elements are in percent weight and δ_c is ferrite in percent volume.

The following equations are taken from Reference 6.

The saturation room temperature (RT) impact energy C_{vsat} (J/cm^2) of cast stainless steel was determined from the chemical composition available from CMTR and provided in Table 3-1.

Since CF8M material toughness values are lower than CF8 material toughness values, calculations are conservatively based on CF8M material.

For CF8M steel with <10% Ni the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10}C_{vsat} = 1.10 + 2.12 \exp(-0.041\phi) \tag{3-1}$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5; \tag{3-2}$$

and from

$$\log_{10}C_{vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \tag{3-3}$$

For CF8M steel with >10% Ni, the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10}C_{vsat} = 1.10 + 2.64 \exp(-0.064\phi) \tag{3-4}$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5 \tag{3-5}$$

and from

$$\log_{10}C_{vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \tag{3-6}$$

The saturation J-R curve at 290°C (554°F), for static-cast CF8M steel is given by

$$J_d = 49\,[C_{vsat}]^{0.41}\,[\Delta a]^n \tag{3-7}$$

$$n = 0.23 + 0.06 \log_{10}[C_{vsat}] \tag{3-8}$$

where J_d is the "deformation J" in kJ/M^2 and Δa is the crack extension in mm.

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[

]a,c,e

a,c,e

N is assumed as 0.05

[1] from Equation 3-1 or 3-4
[2] from Equation 3-3 or 3-6
[3] minimum of items C_{Vsat}[1] and C_{Vsat}[2]

4 LOADS ON THE PUMP CASING NOZZLES

In WCAP-13045 enveloping axial force and moment loadings on the inlet and outlet pump casing nozzles were applied in the three-dimensional finite element analyses of the WOG plant pump casings (Reference 3).

Normal and Faulted Loads

The loads are obtained from the latest primary loop piping stress analysis. Torsional loads are conservatively included.

The moments for the desired loading combinations are calculated by the following equation:

$$M=(M_X{}^2+M_Y{}^2+M_Z{}^2)^{1/2} \tag{4-1}$$

where,

M	=	Moment for Required Loading
M_X	=	X Component of Bending Moment
M_Y	=	Y Component of Bending Moment
M_Z	=	Z Component of Bending Moment

NOTE: X-axis is along the centerline of the nozzle.

Summary and Comparison of Loads

The faulted nozzles loads (i.e., the normal plus safe shutdown earthquake nozzle loads) for the R. E. Ginna pump casings are compared with the screening (i.e., enveloping) faulted loads. The normal loads are compared with screening normal loads for evaluating the loss-of-load condition. The normal and faulted loads utilized here are based on the latest primary loop analyses on record.

In Table 4-1 the normal operating loads obtained, as mentioned above, for R. E. Ginna are compared with the Level C screening nozzle loads (see Table 6-2 of WCAP-13045) which were used for evaluating the loss-of-load upset condition. The R. E. Ginna normal loads at the inlet nozzle and outlet nozzle are bounded by the corresponding Level C screening loads. No additional analysis is required for the Level C loading case. Analysis performed in Reference 3 is conservatively applicable for R. E. Ginna.

The R. E. Ginna faulted loads determined, as mentioned above, are compared in Table 4-2 to the Level A screening loads as defined in WCAP-13045 (see Table 6-2 of WCAP-13045). The R. E. Ginna plant faulted force and moments at the inlet and outlet nozzles are bounded by the faulted screening loads. No additional analysis is required for the Level A loading case. Analysis performed in Reference 3 is conservatively applicable for R. E. Ginna.

Table 4-1 Comparison of the Normal Loads for the Pump Casing Nozzles of R. E. Ginna Nuclear Power Plant with the Screening Level C, Normal Loads

Load	Temperature (°F)	Inlet Nozzle Force (kips)	Inlet Nozzle Moment (in-kips)	Outlet Nozzle Force (kips)	Outlet Nozzle Moment (in-kips)
R. E. Ginna Nuclear Power Plant	543	1760*	5263	1348*	3980
Screening Level C	590	1900	23000	1400	8000

Note: As explained in WCAP-13045, the enveloping stresses were determined for the loss-of-load transient. This was conservatively assumed as the limiting Level C transient.

Table 4-2 Comparison of the Faulted Loads for the Pump Casing Nozzles of R. E. Ginna Nuclear Power Plant with the Level A, Faulted Screening Loads

Load	Temperature (°F)	Inlet Nozzle Force (kips)	Inlet Nozzle Moment (in-kips)	Outlet Nozzle Force (kips)	Outlet Nozzle Moment (in-kips)
R. E. Ginna Nuclear Power Plant	543	1803*	14004	1499*	11962
Screening Level A	550	2000	40000	1800	20000

Notes:

1. Screening loads are taken from Reference 3, Table 6-2.

2. It can be noted from the above Tables that the Ginna forces and moments are significantly lower than the corresponding screening loads.

* Includes force due to the operating pressure of 2250 psi.

5.0 STABILITY EVALUATION

In Section 4.0, it was determined that the normal and faulted forces at the inlet and outlet nozzles of the R. E. Ginna Nuclear Power Plant are bounded by the respective screening loads of WCAP-13045. In this section the stability criteria are investigated for the R. E. Ginna Nuclear Power Plant primary loop pump casings.

As explained in Section 10.0 of WCAP-13045, a postulated flaw is stable if either:

1. $J_{applied} < J_{Ic}$ or
2. If $J_{applied} > J_{Ic}$ then $T_{applied} < T_{material}$ and $J_{applied} < J_{max}$

The limiting material toughness values are listed in Section 3.0 of this report. They are:

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6 FATIGUE CRACK GROWTH ASSESSMENT

6.1 INTRODUCTION

Fatigue crack growth cracks are postulated at various locations in the pump casings. Such postulated cracks would be subject to the various cyclic conditions the pump casing experience. Thus, the sensitivity to cyclic loadings of postulated cracks in the pump casings was evaluated as a generic fatigue crack growth analysis for pump casing Model 93 in Section 12 of Reference 3.

The highest stressed location was chosen for the fatigue crack growth. This region is at Flaw 5-93. The postulated flaws are at the outlet nozzle knuckle in the plane of the weld. The stress contours for Level A loads are given in Figure 8-8 of Reference 3 and typify this location.

The generic transients considered for the fatigue crack growth are given in Table 12-2 of Reference 3. The fatigue crack growth results taken from Table 12-2 of Reference 3 are also given in Table 6-1. The maximum acceptable flaw size (0.30-in.) is seen to increase by less than [

]a,c,e

6.2 DISCUSSION AND CONCLUSIONS

The highest stressed location in the Models 93-pump casing has been evaluated for fatigue crack growth. The crack growth observed is bounding for other less severely stressed locations. For 40 year plant life, postulated crack depths initially well in excess of the maximum ASME code allowable remain well less than the flaw sizes shown to be stable in Reference 3.

It is concluded that any reasonably sized flaws in the pump casings will exhibit only minimal crack extension during service life of 40 years, such flaws remaining well below the flaw sizes shown to be stable.

The transients and cycles of R. E. Ginna Nuclear Power Plant for 60 year plant life are the same as those of 40 year plant life and therefore the fatigue crack growth results shown in Table 6-1 for 40-year are also applicable for 60-year.

Table 6-1 Fatigue Crack Growth for Postulated Flaws in the Outlet Nozzle Knuckle Region of the Model 93 Pump Casings

Initial Crack Depth (in.)	Crack Depth (in.) at End of Year			
	10	20	30	40 and 60
[				
				]a,c,e

1 The maximum acceptable depth of a flaw per Table IWB 3518-2 of Section XI of the ASME Code (1989 Edition).

7 DISCUSSION AND CONCLUSIONS

This report provides an assessment of the primary loop pump casings of R. E. Ginna Nuclear Power Plant primary loop to the conditions of Item (d) of ASME Code Case N-481 (see Appendix A).

This evaluation considers actual R. E. Ginna Nuclear Power Plant primary loop pump casings fracture toughness values. Thus Item (d) (1) is satisfied.

Stress analyses of a representative primary loop pump casing are presented in WCAP-13045. This satisfies Item (d) (2).

The operating history of Westinghouse design primary loop pumps is reviewed in Section 2.0 of WCAP-13045. This satisfies Item (d) (3). Flaws are postulated in the pump casings as described in Section 9.0 of WCAP-13045 satisfying Item (d) (4). One-quarter thickness reference flaws with a six-to-one aspect ratio are postulated in WCAP-13045 consistent with Item (d) (5).

Comparisons of the loads of the R. E. Ginna Nuclear Power Plant primary loop pump casings with the screening loads of WCAP-13045 are presented in this report. The stability of the flaws postulated in the R. E. Ginna Nuclear Power Plant primary loop pump casings are established by evaluating the resulting $J_{applied}$ and $T_{applied}$ against the fracture toughness values noted in the discussion of Item (d) (1) (See Section 5.0). This satisfies Item (d) (6).

The preservice fracture toughness of cast stainless steels is very high. Thermal aging causes a reduction in the toughness. The effect of thermal aging has been evaluated in Section 3.0 of this report and Appendix A of WCAP-13045. No other mechanism is known to degrade the properties of the pump casings during service. Item (d) (7) is so satisfied.

It is concluded that the primary loop pump casings of R. E. Ginna Nuclear Power Plant are in compliance with Item (d) of ASME Code Case N-481 for 40-year and 60-year (as a part of the License Renewal program) plant life.

8 REFERENCES

1. WCAP-7211, Revision 4, "Proprietary Information and Intellectual Proprietary Management Policies and procedures," January 2001.

2. Case N-481: "Alternate Examination Requirements for Cast Austenitic Pump Casings," Section XI, Division 1, Cases of ASME Boiler and Pressure Vessel Code, Approval Date: March 5, 1990.

3. F. J. Witt and J. F. Petsche, Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems, WCAP-13045, September 1991 (Westinghouse Proprietary Class 2).

4. WCAP-13606, Revision 1, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of the Robert E. Ginna Nuclear Power Plant," April 1993..

5. O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U. S. Nuclear Regulatory Commission, Washington, DC, May 1994.

6. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG-CR-4513, Revision 1, U. S. Nuclear regulatory Commission, Washington, DC, August 1994.

7. "Flaw Evaluation of Thermally aged Cast Stainless Steel in Light-Water Reactor Applications," Lee, S.; Kuo, P. T.; Wichman, K.; Chopra, O.; Published in International Journal of Pressure Vessel and Piping, June 1997.

APPENDIX A
CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: March 5, 1990
See Numerical Index for expiration and any reaffirmation dates.

Case N-481
Alternate Examination Requirements for
Cast Austenitic Pump Casings
Section XI, Division 1

Inquiry: When conducting examination of cast austenitic pump casings in accordance with Section XI, Division 1, what examinations may be performed in lieu of the volumetric examinations specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

Reply: It is the opinion of the Committee that the following requirements shall be met in lieu of performing the volumetric examination specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

(a) Perform a VT-2 visual examination of the exterior of all pumps during the hydrostatic pressure test required by Table IWB-2500-1, Category B-P.

(b) Perform a VT-1 visual examination of the external surfaces of the weld of one pump casing.

(c) Perform a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance.

(d) Perform an evaluation to demonstrate the safety and serviceability of the pump casing. The evaluation shall include the following:

(1) evaluating material properties, including fracture toughness values;

(2) performing a stress analysis of the pump casing;

(3) reviewing the operating history of the pump;

(4) selecting locations for postulating flaws;

(5) postulating one-quarter thickness reference flaw with a length six times its depth;

(6) establishing the stability of the selected flaw under the governing stress conditions;

(7) considering thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service.

(e) A report of this evaluation shall be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review.