Mechanical-Property Degradation of Cast Stainless Steel Components from the Shippingport Reactor

ANL/CP--74953

DE92 006967

O. K. Chopra

Materials and Components Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439 USA

OCT 5 1992

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

The submitted manuscript has been authored by a contractor of the U.S. Government under contract No. W-31-109-ENG-38. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

October 1991

Presented at the Nineteenth Water Reactor Safety Information Meeting, October 28-30, 1991, Bethesda, MD.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED.

Mechanical-Property Degradation of Cast Stainless Steel Components from the Shippingport Reactor

O. K. Chopra

Materials and Components Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Abstract

The mechanical properties of cast stainless steels from the Shippingport reactor have been characterized. Baseline properties for unaged materials were obtained from tests on either recovery-annealed material or material from a cooler region of the component. The materials exhibited modest decrease in impact energy and fracture toughness and a small increase in tensile strength. The fracture toughness J-R curve, J_{IC} value, tensile flow stress, and Charpy-impact energy of the materials showed very good agreement with estimations based on accelerated laboratory aging studies. The kinetics of thermal embrittlement and degree of embrittlement at saturation, i.e., the minimum impact energy that would be achieved after long-term aging, were established from materials that were aged further in the laboratory at temperatures between 320 and 400°C. The results showed very good agreement with estimates; the activation energies ranged from 125 to 250 kJ/mole and the minimum room-temperature impact energy was $>75 \text{ J/cm}^2$. The estimated impact energy and fracture toughness J-R curve for materials from the Ringhals reactor hot and crossover-leg elbows are also presented.

1 Introduction

Cast duplex stainless steels used in LWR systems for primary pressure-boundary components such as valve bodies, pump casings, and primary coolant piping are susceptible to thermal embrittlement after extended service at reactor operating temperatures, i.e., 280–320°C (536–608°F). Aging of cast stainless steels at these temperatures leads to increased hardness and tensile strength and decreased ductility, impact strength, and fracture toughness. Most studies of thermal embrittlement of cast stainless steels involve simulation of end-of-life reactor conditions by accelerated aging at higher temperatures, viz., 400°C (752°F), because the time period for operation of power plants ($\approx 40 \text{ y}$) is far longer than can generally be considered in laboratory studies. Thus, estimates of mechanical-property degradation suffered by cast stainless steel components during service are based on an Arrhenius extrapolation of high-temperature data to reactor operating conditions.

Several laboratory studies have investigated thermal embrittlement of cast stainless steels under LWR operating conditions.^{1–12} A procedure and correlations have been developed for estimating fracture toughness, tensile, and Charpy-impact properties of cast stainless steel components during thermal aging.^{13,14} Because the embrittlement mechanisms and kinetics are complex, microstructural studies and mechanical testing of actual component materials that have completed long in-reactor service are necessary, to ensure that the

the mechanisms observed in accelerated aging experiments are the same as those occurring in reactors. Cast stainless steels from the decommissioned Shippingport reactor offer a unique opportunity to validate and benchmark the laboratory studies.

The objectives of this paper are to characterize the mechanical-property degradation of cast stainless steel components from the Shippingport reactor and compare the results with estimates from accelerated laboratory aging studies. Cast stainless steel materials were obtained from four cold-leg check valves, two hot-leg main shutoff valves, and two pump volutes of the Shippingport reactor. One of the volutes is a "spare" that had seen service only during the first core loading, whereas the other was in service for the entire life of the plant. The actual time-at-temperature for the materials was ≈ 13 y at $\approx 281^\circ\text{C}$ (538°F) for the hot-leg components and $\approx 264^\circ\text{C}$ (507°F) for the cold-leg components. The components were in a hot stand-by condition of $\approx 204^\circ\text{C}$ (400°F) for an additional ≈ 2 y. Service-aged material was also obtained from the recirculating-pump cover assembly of the KRB reactor, which was in service in Gundremmingen, Germany, for ≈ 8 y at 284°C (543°F).

The chemical composition, hardness, and ferrite content and distribution for the cast materials are given in Table 1. All materials are CF-8 grade cast stainless steel with ferrite contents in the range of 2-16% for the Shippingport components and 34% for the KRB pump cover plate. Results from metallurgical characterization of the various Shippingport materials have been presented earlier.¹⁵ The materials show a lacy ferrite morphology with an average ferrite spacing in the range of 150-300 μm . The check valve materials contain phase-boundary carbides, which most likely formed during production heat treatment of

Table 1. Chemical composition, ferrite morphology, and hardness of cast stainless steel components from the Shippingport, KRB, and Ringhals reactors.

Mater. ID ^a	Composition (wt.%)										Ferrite		Ferrite Spacing (μm)	Hard- ness (R _P)
	C	N	Si	Mn	P	S	Ni	Cr	Mo	Cu	Calc. (%)	Meas. (%)		
<u>Cold Leg Check Valve^b</u>														
CA4	0.056	0.041	1.45	1.10	0.018	0.009	8.84	20.26	0.01	0.07	10.8	10.9	157	79.8
CA7	0.058	0.041	1.43	1.09	0.018	0.009	8.72	20.22	0.01	0.07	10.9	10.0	148	78.6
CB7	0.052	0.053	1.36	1.07	0.018	0.011	8.85	19.12	0.02	0.06	5.9	3.2	296	75.0
<u>Hot Leg Main Shutoff Valve^b</u>														
MA1	0.052	0.049	0.22	0.72	0.039	0.013	10.50	20.74	0.24	0.13	5.2	9.5	217	76.9
MA9	0.052	0.051	0.24	0.72	0.041	0.011	10.54	20.78	0.24	0.13	5.1	10.0	245	77.6
MB2	0.042	0.073	0.51	0.72	0.043	0.017	10.77	19.74	0.19	0.12	2.6	1.9	-	74.2
<u>Pump Volute^c</u>														
VR	0.046	0.049	1.14	0.50	0.027	0.017	9.56	20.79	0.04	0.07	9.8	16.2	181	82.9
PV	0.108	0.027	0.89	1.11	0.032	0.008	9.30	19.83	0.38	0.25	4.1	13.0	-	-
<u>KRB Pump Cover Plate^d</u>														
KRB	0.062	0.038	1.17	0.31	-	-	8.03	21.99	0.17	-	27.7	34.0	-	-
<u>Ringhals Reactor Elbows^e</u>														
H	0.037	0.044	1.03	0.77	0.022	0.008	10.60	20.00	2.09	0.17	13.0	20.1	-	-
C	0.039	0.037	1.11	0.82	0.020	0.012	10.50	19.60	2.08	0.08	12.3	19.8	-	-

^a For the valves, the second digit indicates the loop where the valve was located and the number designates the segment of the component from which the material was removed (segments 1, 2, and 7 are from the top of the valve, segment 4 is from the bottom, and segment 9 is from a cooler region).

^b In service for ≈ 13 y at 264°C for cold leg and at 281°C for hot leg.

^c Spare pump volute VR in service only during initial core loading and PV in service for ≈ 13 y at 264°C .

^d In service for ≈ 8 y at 284°C .

^e In service for ≈ 8 y at 325°C for hot leg and at 291°C for crossover leg, and a hot stand-by for 2.3 y at 303°C for hot leg and at 274°C for crossover leg.

the casting. Also, the check-valve material showed phase boundary migration, and the original phase boundaries were decorated with carbides.

Microstructural examination indicates that the mechanism of low-temperature embrittlement of the cast materials is the same as that of the laboratory-aged materials.^{16,17} All materials showed spinodal decomposition of the ferrite to form a chromium-rich α' phase. In addition, the check-valve materials contained a nickel- and silicon-rich G phase in the ferrite and $M_{23}C_6$ carbides at the austenite/ferrite phase boundary. An unexpected microstructural feature, i.e., σ phase precipitates on slip bands and stacking faults, was also observed in the austenite of the check-valve material. Precipitation of σ phase generally occurs at temperatures $>550^\circ\text{C}$ ($>1022^\circ\text{F}$). The presence of σ phase and phase-boundary migration indicate significant differences between the production heat treatment of the check valves and that of the other materials.

The mechanical-property degradation of cast stainless steel (CF-8M) elbows from the Ringhals reactor hot leg and crossover leg was also assessed and compared with experimental data.¹⁸ The chemical composition, ferrite content, and service conditions for the materials are given in Table 1.

2 Mechanical Properties

Specimens for Charpy-impact, tensile, and fracture toughness J-R curve tests were obtained from different locations across the thickness of the various components. All specimens were in the LC orientation.* Impact tests were conducted on standard Charpy V-notch specimens machined according to ASTM Specification E 23. A Dynatup Model 8000A drop-weight impact machine with an instrumented tup and data readout system was used for the tests. Tensile tests were performed on cylindrical specimens 5 mm in diameter, with a gauge length of 20 mm. The tests were conducted at an initial strain rate of $4 \times 10^{-4} \text{ s}^{-1}$. The J-R curve tests were conducted according to ASTM Specifications E 813-85 and E 1152-87. Compact-tension specimens, 25.4-mm thick, were used for the tests.

The baseline mechanical properties for the as-cast materials must be known to establish the thermal-aging effects during reactor service. Microstructural and annealing studies^{2,15-17,19} on laboratory- and reactor-aged materials indicate that mechanical properties of unaged material can be determined from recovery-annealed material, i.e., embrittled material that has been annealed for 1 h at 550°C and then water-quenched. To obtain baseline properties, Charpy-impact tests were also conducted on material from a cooler region of the main shutoff valve. Charpy transition curves for MA9 material and recovery-annealed MA9 and MA1 material are shown in Fig. 1. These materials are from the Loop A main shutoff valve, although MA9 is from a cooler region of the valve. The results indicate that MA9 material suffered little or no thermal-aging embrittlement, i.e., annealing had no effect on the transition curves. The results for annealed MA1 material also show good agreement on the transition curves. The results for annealed MA1 also show good agreement with the transition curve for MA9. The upper-shelf energy (USE) for both materials is not constant but decreases as temperature increases. The average impact energies at room temperature

* The first digit represents the direction normal to the plane of a crack and the second digit indicates the direction of crack propagation: L = longitudinal and C = circumferential.

Figure 1.
Effect of annealing on Charpy transition curve for cast material from the hot-leg main shutoff valve. Material MA9 is from a cooler region of the valve. δ_c and δ_m are calculated and measured ferrite contents, respectively.

and at 290°C, respectively, are 356 and 253 J/cm² for MA9, and 320 and 254 J/cm² for annealed MA1. The Charpy data were fitted with a hyperbolic tangent function of the form

$$C_V = K_0 + B[1 + \tanh [(T - C)/D]] \quad (1)$$

where K_0 is the lower-shelf energy, T is the test temperature in °C, B is half the distance between upper- and lower-shelf energy, C is the mid-shelf Charpy transition temperature (CTT) in °C, and D is the half-width of the transition region. The best-fit curves for MA9, with or without annealing, and for annealed MA1 indicate that the latter is marginally weaker; the CTT of MA1 is ≈10°C higher and the average USE ≈30 J/cm² lower. Such differences in impact energy are most likely due to minor variations in composition and structure of the materials from different locations of the casting. The Charpy data for MA9 and annealed MA1 may be represented by a single best-fit transition curve, as shown in Fig. 1.

Charpy Impact Energy

Charpy transition curves for the various cast materials from the Shippingport reactor are shown in Figs. 2-4. The Charpy data were fitted with the hyperbolic-tangent expression given in Eq. 1; the values of the constants for the various materials are given in Table 2. The results indicate that the room-temperature impact energy of the materials is relatively high and the mid-shelf CTT, i.e., constant C in Eq. 1, is very low. The check valve materials CA4 and CB7 are weaker than MA1 and PV, e.g., the mid-shelf CTT is ≈100°C higher for CA4 and CB7. The higher CTTs are due to the presence of phase-boundary carbides in the check-valve materials. The carbides weaken the phase boundaries and thus provide an easy path for fracture.

The decrease in impact strength from ≈13 y of service at reactor temperatures is minimal for the materials. The room-temperature impact energy of PV, MA1, and CA4 materials is decreased by ≈90, 70, and 40 J/cm², respectively. The large difference in USE for the unaged and service-aged materials from Row 1 of MA1 (Fig. 2), is not due to thermal aging. The inner 15-mm region of the MA1 valve body contains a high density of inclusions/flaws and is inherently weak. The inner surface of all the valves contained repair welds. No significant difference was observed in the chemical composition or ferrite content of the material across the thickness of the valve body.

Table 2. Values of constants in Eq. 1 for Charpy transition curve of CF-8 cast stainless steels from the Shippingport reactor and KRB pump cover plate

Material ID	Service Condition		Constants			
	Temp. (°C)	Time (y)	K_0 (J/cm ²)	B (J/cm ²)	C (°C)	D (°C)
<u>Cold-Leg Check Valves:</u>						
CA4	Annealed	-	25	98.6	-37.0	97.9
CA4	264	13	25	79.2	-20.1	81.8
CB7	264	13	76	108.8	6.0	65.2
<u>Hot-Leg Main Shutoff Valve:</u>						
MA9 ^a	Annealed	-	96	112.0	-116.3	54.1
MA9	<200	13	83	110.1	-110.7	48.3
MA1/23 ^a	Annealed	-	96	112.0	-116.3	54.1
MA1/23 ^b	281	13	73	87.6	-114.2	29.8
MA1/1 ^c	281	13	69	63.7	-137.0	38.6
<u>Pump Volutes:</u>						
PV	Annealed	-	150	116.2	-151.9	109.7
PV	264	13	75	109.4	-141.9	49.5
VR	Unaged	-	61	88.1	-112.4	38.5
<u>Pump Cover Plate^d</u>						
KRB	Annealed	-	8	161.9	-16.5	87.2
KRB	284	8	8	119.7	36.8	83.2

^a Determined from combined data for MA9 and annealed MA9 and MA1.

^b Material from Rows 2 & 3, which corresponds to 15- to 45-mm region of the wall.

^c Material from Row 1, which corresponds to inner 15-mm region of the wall.

^d Obtained from the KRB reactor in Gundremmingen, Germany.

Figure 2. Charpy transition curves for hot-leg main shutoff valve after 13 y service at 281°C. Row 1 represents inner 15-mm region and rows 2 and 3 represent 15- to 50-mm region of the valve body. Results from recovery-annealed MA1 are shown as the baseline Charpy transition curve.

Figure 3. Charpy transition curves for cold-leg check valves from Loops A and B after 13 y of service at 264°C. Baseline transition curve for CA4 is represented by results for recovery-annealed material.

Figure 4. Charpy transition curves for service-aged (13 y of service at 264°C) and spare pump volutes. Baseline transition curve for PV is represented by results for recovery-annealed material.

Tensile Properties

Tensile tests were conducted at room temperature and at 290°C on CA4, PV, MA1, and MA9 materials. Tensile properties were also estimated from the instrumented Charpy-impact test data. For a Charpy specimen, yield stress is given by

$$\sigma_y = 1.50P_y B/Wb^2, \quad (2)$$

and ultimate stress by

$$\sigma_u = 2.28P_m B/Wb^2, \quad (3)$$

Figure 5. Yield and ultimate stress values (estimated from Charpy-impact data and obtained from tensile tests) for cold-leg check valve and pump volute, and estimated tensile stresses of recovery-annealed materials.

Figure 6. Yield and ultimate stress values (estimated from Charpy-impact data and obtained from tensile tests) for hot-leg main valve, and estimated tensile stresses for recovery-annealed materials. Material MA9 is from a cooler region of the valve.

where P_y and P_m are the yield and maximum loads obtained from the load-time traces of the Charpy test, W is the specimen width, B is the specimen thickness, and b is the uncracked ligament.²⁰ The estimated values of yield and ultimate stress, the values obtained from tensile tests, and estimated tensile stresses for recovery-annealed materials are shown in Figs. 5 and 6.

The estimated tensile properties are in good agreement with the measured values. The tensile strength of CA4, PV, and MA1 materials is comparable. The results show that thermal aging during reactor service had no effect on yield stress and that the increase in ultimate stress is minimal for all materials. Two specimens of MA1 (Fig. 6) show low ultimate strength (and also poor ductility). These specimens were obtained from the inner-15-mm

region of the valve body. The poor tensile properties are caused by inclusions in the material. As discussed above, the room-temperature impact energy of Row 1 specimens is also low, e.g., $\approx 177 \pm 33$ J/cm², compared to $\approx 299 \pm 33$ J/cm² for specimens from other regions of the valve body.

3 Estimation of Mechanical Properties

Charpy-Impact Energy

A procedure and correlations have been presented* for predicting Charpy-impact energy, tensile flow stress, and fracture toughness J-R curves of aged cast stainless steels from the chemical composition of the steel. Embrittlement of cast stainless steels is characterized in terms of room-temperature Charpy-impact energy. The extent or degree of embrittlement at "saturation," i.e., the minimum impact energy that would be achieved for the material after long-term aging, is described in terms of a material parameter Φ , which, for CF-3 and CF-8 steels, is expressed as:

$$\Phi = \delta_c(\text{Cr} + \text{Si})(\text{C} + 0.4\text{N}) \quad (4a)$$

and for CF-8M steels as:

$$\Phi = \delta_c(\text{Ni} + \text{Si} + \text{Mn})^2(\text{C} + 0.4\text{N})/5, \quad (4b)$$

where the ferrite content δ_c is in % and chemical composition of the steel is in wt.%. The ferrite content is calculated from Hull's equivalent factors;²¹ the values for the various cast materials are given in Table 1. The saturation room-temperature impact energy Cv_{sat} (J/cm²) for CF-3 and CF-8 steels is given by:

$$\log_{10} Cv_{\text{sat}} = 1.15 + 1.36 \exp(-0.025\Phi), \quad (5a)$$

and for CF-8M steels with $>10\%$ Ni by

$$\log_{10} Cv_{\text{sat}} = 1.10 + 2.64 \exp(-0.064\Phi), \quad (5b)$$

The room-temperature saturation impact energy is also estimated from the chemical composition of the steel. For CF-3 and CF-8 steels, Cv_{sat} (J/cm²) is given by:

$$\log_{10} Cv_{\text{sat}} = 5.64 - 0.006\delta_c - 0.185\text{Cr} + 0.273\text{Mo} - 0.204\text{Si} + 0.044\text{Ni} - 2.12(\text{C} + 0.4\text{N}), \quad (6a)$$

and for CF-8M steels by

$$\log_{10} Cv_{\text{sat}} = 7.28 - 0.011\delta_c - 0.185\text{Cr} - 0.369\text{Mo} - 0.451\text{Si} - 0.007\text{Ni} - 4.71(\text{C} + 0.4\text{N}), \quad (6b)$$

The saturation impact energy for a specific cast stainless steel is determined by both the methods given in Eqs. 4-6; and the lower value is then used for estimating mechanical properties.

* "Estimation of Mechanical Properties of Cast Stainless Steels during Thermal Aging in LWR Systems," this conference.

Room-temperature impact energy as a function of time and temperature of aging is estimated from the room-temperature saturation impact energy Cv_{sat} and kinetics of embrittlement. The decrease in room-temperature impact energy Cv (J/cm²) with time is expressed as:

$$\log_{10} Cv = \log_{10} Cv_{sat} + \beta \{1 - \tanh [(P' - \theta)/\alpha]\}, \quad (7)$$

where β is half the maximum change in $\log Cv$, θ is the log of the time at 400°C to achieve β reduction in impact energy at 400°C, and α is a shape factor. The aging parameter P' is the log of the aging time for a specific degree of embrittlement and is defined by

$$P' = \log_{10} t - \frac{10000Q}{19.143 [T_s + 273] - 673}, \quad (8)$$

where Q is the activation energy (kJ/mole) for thermal embrittlement and t and T_s are the time (h) and temperature (°C) of aging, respectively. The activation energy Q (kJ/mole) is also determined from the chemical composition of the steel. For CF-3 and CF-8 steels:

$$Q = 10 [74.52 - 7.20\theta - 3.46Si - 1.78Cr + 148N - 61C] \quad (9a)$$

and for CF-8M steels:

$$Q = 10 [74.52 - 7.20\theta - 3.46Si - 1.78Cr - 4.35Mn + 23N], \quad (9b)$$

where the constant θ is defined in Eq. (7). The constants β and α in Eq. (7) can be determined from the initial room-temperature impact energy of the material Cv_{int} and the saturation room-temperature impact energy Cv_{sat} . Thus,

$$\beta = (\log_{10} Cv_{int} - \log_{10} Cv_{sat})/2, \quad (10)$$

and

$$\alpha = -0.585 + 0.795 \log_{10} Cv_{sat}. \quad (11)$$

For a specific cast stainless steel, the values of room-temperature impact energy as a function of time and temperature of reactor service and the minimum impact energy that would be achieved for the material after long-term aging (i.e., saturation value) can be estimated from Eqs. 4-11. The information required for the estimations includes the chemical composition, initial impact energy of the unaged material, and the constant θ . However, parametric studies indicate that the aging response at 280-330°C is relatively insensitive to the value of θ . Varying θ between 2.1 and 3.6 results in almost identical aging behavior at 300°C and differences in aging behavior at 280-330°C are minimal. A value of 2.9 is assumed for the constant θ in Eqs. 7 and 9 when estimating impact energy of cast stainless steel components in service at 280-330°C.

Room-temperature impact energy of the various service-aged materials was estimated from Eqs. 4-11. The initial impact energy of the unaged materials was determined from the data for recovery-annealed material or material from a cooler region of the casting. Some materials were aged further in the laboratory at 320, 350, and 400°C (608, 662, and 752°F) to obtain an accurate value of θ and to validate the estimates of the saturation impact energy Cv_{sat} and activation energy for embrittlement of the materials.

Figure 7. Variation of estimated room-temperature Charpy-impact energy with service time for Loop A hot-leg main valve materials MA1 and MA9. Material MA9 is from a cooler region of the valve. δ_c is calculated ferrite content.

Figure 8. Variations of estimated room-temperature Charpy-impact energy with service time for Loop A cold-leg check valve CA4 and pump volute PV. δ_c is calculated ferrite content.

The change in estimated Charpy-impact energy with aging time at temperatures between reactor service temperature for the Shippingport and KRB materials and 400°C is shown in Figs. 7-9. The high-temperature aging data for CA4 and MA1 materials represent service-aged material that was aged further in the laboratory at 350 and 400°C. Aging times were adjusted to include the effect of aging at reactor temperature. The high-temperature aging data for the KRB pump cover plate were obtained on recovery-annealed material.

The impact energies estimated with experimental θ show good agreement with the measured impact energies at all temperatures; those estimated with $\theta = 2.9$ show good agreement at temperatures $\leq 320^\circ\text{C}$. As mentioned above, estimations based on $\theta = 2.9$ are valid only for service temperatures between 280 and 330°C. For higher temperatures, the estimated values would be nonconservative for materials that have a θ value < 2.9 , e.g., KRB pump cover plate and VR material (Fig. 9). A θ value of 2.5 would give a conservative estimate of impact energy for all materials at 330-360°C service temperature.

Figure 9. Variations of estimated room-temperature Charpy impact energy with service time for the KRB pump cover plate and Shippingport spare pump volute VR. δ_0 is calculated ferrite content.

The impact energy for main valve MA1 was estimated from the compositions of MA1 and MA9 materials; the differences in the compositions of the two materials are minor. Figure 7 shows that, although the aging behavior at 400°C and the kinetics of embrittlement for MA1 and MA9 are significantly different, the estimates based on MA1 and MA9 agree well with the observed values for ≈ 13 y of service at 281°C. The aging behavior estimated from MA9 is slightly slower than that estimated from MA1.

The predicted minimum saturation room-temperature impact energies also are in very good agreement with the experimental data. The measured impact energies for VR, MA9, and KRB materials aged at 400°C achieve saturation at the predicted values.

Fracture Toughness

Thermal aging decreases the fracture toughness of cast stainless steels at both room temperature and reactor temperature. The fracture toughness J-R curve for a specific cast stainless steel can be estimated from its room-temperature impact energy C_V (J/cm²). The J-R curve is expressed by the power-law relation $J_R = C\Delta a^n$, where J_R is deformation J (kJ/m²) per ASTM Specifications E 813-85 and E 1152-87, Δa is crack extension (mm), and C and n are constants. At room temperature, the J-R curve for static-cast CF-8 steels is given by

$$J_R = 49[C_V]^{0.52}[\Delta a]^n, \quad (12a)$$

and for static-cast CF-8M steels by

$$J_R = 16[C_V]^{0.67}[\Delta a]^n. \quad (12b)$$

At 290-320°C, the J-R curve for static cast CF-8 steels is given by

$$J_R = 102[C_V]^{0.28}[\Delta a]^n. \quad (13a)$$

and for static-cast CF-8M steels by

$$J_R = 49[Cv]^{0.41}[\Delta a]^n \quad (13b)$$

At room temperature, the exponent n for static- or centrifugally cast CF-8 steels is given by

$$n = 0.22 + 0.139 \log_{10} Cv. \quad (14a)$$

and for static-cast CF-8M steels by

$$n = 0.25 + 0.077 \log_{10} Cv. \quad (14b)$$

At 290-320°C, the exponent n for static- or centrifugally cast steels is given by

$$n = 0.22 + 0.074 \log_{10} Cv. \quad (15a)$$

and for static-cast CF-8M steels by

$$n = 0.23 + 0.057 \log_{10} Cv. \quad (15b)$$

The estimated and experimental fracture toughness J - R curves at room temperature and at 290°C for the CA4, MA1, and PV materials and KRB pump cover plate are shown in Figs. 10 and 11. All materials exhibit relatively high fracture toughness. The estimated J - R curves for CA4 and MA1 show good agreement with the experimental results and those for PV and KRB materials are 30-50% lower. The lower values for the PV and KRB materials are due to a conservative estimate of the kinetics of embrittlement, i.e., the correlations predict a faster decrease in impact energy than that observed during service, i.e., the estimated room-temperature impact energy is $\approx 40\%$ lower than the average experimental value.

Figure 10. Estimated and measured fracture toughness J - R curve at room temperature and 290°C for the Shippingport cold-leg check valve and hot-leg main valve

Tensile Properties

Thermal aging of cast stainless steels increases their tensile strength, particularly their ultimate stress. The tensile strength of the unaged materials from the Shippingport and KRB reactors were determined from tensile tests or estimated from instrumented Charpy-impact tests on recovery-annealed materials. The materials show little or no increase in tensile stress. The increase in flow stress of aged cast stainless steels is estimated from a

Figure 11. Estimated and measured fracture toughness J-R curve at room temperature and 290°C for the Shippingport cold-leg pump volute and the KRB pump cover plate

correlation between the ratio of tensile flow stress of aged and unaged material and a normalized aging parameter. Flow stress is characterized as the mean of the 0.2% yield and ultimate stresses, and the aging parameter P (defined in Eq. 8) is normalized with respect to a θ value of 2.9. The flow stress ratio $R = (\sigma_{\text{aged}}/\sigma_{\text{unaged}})$ is given by

$$R = a + b(P - \theta + 2.9). \quad (16)$$

Equation 16 is valid for ferrite contents $>7\%$ and R values between 1 and a constant c . Values of the constants a , b , and c for different grades of steel and test temperatures are given in Table 3. The tensile flow stresses for the Shippingport and KRB materials were estimated from Eq. 16 with a value of 2.9 for θ . The results, given in Table 4, show good agreement with the measured values.

Fracture toughness J_{IC} values for service-aged materials can be determined from the estimated J-R curve and flow stress, and are also given in Table 4. The estimated J_{IC} shows very good agreement with the measured value for CA4 and is conservative for MA1 and PV.

Table 3. Values of the constants in Eq. 16 for estimating tensile flow stress of aged cast stainless steels

Grade	Room Temp.			290-320°C		
	a	b	c	a	b	c
CF-3	0.94	0.047	1.10	0.89	0.059	1.08
CF-8	0.90	0.074	1.16	0.87	0.088	1.14
CF-8M	0.80	0.101	1.19	0.71	0.143	1.24

4 Ringhals Reactor Elbows

Investigation of the hot- and crossover-leg elbows from the Ringhals reactor indicated significant degradation of impact strength and fracture toughness of the hot-leg elbow after 15 y of service at 325°C, whereas the crossover-leg elbow, in service at 291°C, showed only

Table 4. Measured and estimated tensile flow stress and J_{IC} values for service-aged cast stainless steels

	Temp.	Flow Stress (MPa)		J _{IC} (kJ/m ²)	
Material ID	(°C)	Observed ^a	Estimated	Observed	Estimated
<u>Shippingport Components</u>					
CA4	25	377 (377)	397	476	485
	290	251 (246)	259	361	331
PV	25	370 (362)	395	1509	692
	290	266 (269)	295	858	373
MA1	25	345 (345)	354	1407	809
	290	237 (233)	238	739	417
<u>KRB Pump Cover Plate</u>					
KRB	25	428 (428)	475	263, 396	297
	290	329 (294)	342	681	249
<u>Ringhals Elbows</u>					
Hot	25	424 (399)	469	250, 330 195, 150	165
	350	315 (292)	362		190
Cross-over	25	392 (369)	409	960, 525 960, 600	247
	350	277 (290)	333		243

^a The values within parentheses represent unaged material.

moderate degradation.¹⁸ The mechanical properties of the Ringhals elbows were estimated from Eqs. 4-16 with the correlations for CF-8M steel containing >10% Ni. Information on the chemical composition and impact energy of the unaged materials was used in the estimations; θ was assumed to be 2.9.

The experimental data and estimated decrease in impact energy for hot- and crossover-leg elbows during service at 325 and 291°C, respectively, are shown in Fig. 12. The estimated value of 67 J/cm² for the hot-leg elbow is marginally higher than the measured average values of 45 J/cm² (equivalent Charpy V-notch impact energy converted from U-notch value) and 50 J/cm² (from Charpy V-notch specimens). The estimated 112 J/cm² impact energy for the crossover-leg agrees well with 107 J/cm² measured from U-notch specimens and is significantly lower than the 177 J/cm² obtained from V-notch specimens. The difference between the V- and U-notch impact energy for the crossover-leg elbow is most likely due to a significant variation in the ferrite content of the material. The saturation impact energies for hot- and crossover-leg elbows are estimated to be 56 and 67 J/cm², respectively.

Fracture toughness J-R curves can be estimated from the impact energy. Room temperature J-R curve for hot- and crossover-leg elbows after ≈15 y of service are shown in Fig. 13. Only the experimental J_{IC} values (not the complete J-R curve) have been reported for these materials.¹⁸ The estimated tensile flow stress and J_{IC} at room temperature and at 290°C for the Ringhals elbows are given in Table 4. The estimated flow stresses are in good agreement with the measured values. The J_{IC} for the hot-leg elbow also is comparable to the measured value whereas that for the crossover-leg elbow is 50-70% lower. As mentioned above, the ferrite content of the crossover-leg elbow varies significantly. Furthermore, the correlations do not consider the effect of microstructural differences, and may be conservative for some materials.

Figure 12.
Estimated and experimentally observed room-temperature Charpy-impact energy for the Ringhals hot- and crossover-leg elbow.

Figure 13. Estimated fracture toughness J - R curve at room temperature for the Ringhals hot- and crossover-leg elbows after ≈ 15 y of service

5 Conclusions

Charpy-impact, tensile, and fracture toughness properties of several cast stainless steel materials from the Shippingport, KRB, and Ringhals reactors have been characterized. Baseline mechanical properties for as-cast material were determined from tests on either recovery-annealed material, i.e., material that had been annealed for 1 h at 550°C and then water quenched, or on material from a cooler region of the component. The Shippingport materials exhibited modest degradation of mechanical properties. The room-temperature impact energy was relatively high, >120 J/cm² (>70 ft-lb). Check-valve materials were weaker than main valve materials because of the presence of phase-boundary carbides. The CTT for the check valves was $\approx 100^\circ\text{C}$ higher than that for the main shutoff valves or pump volute. The results show good agreement with estimations based on accelerated laboratory aging studies. The procedure and correlations developed at ANL for estimating thermal aging degradation of cast stainless steels, predict accurate or slightly conservative values for Charpy-impact energy, tensile flow stress, fracture toughness J - R curve, and J_{IC} of the Shippingport and KRB materials. The correlations successfully predict the mechanical properties of the Ringhals reactor hot- and crossover-leg elbows after service of ≈ 15 y.

Acknowledgments

This work was supported by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. The author is grateful to A. Sather, T. M. Galvin, L. Y. Bush, G. M. Dragel, and W. F. Burke for experimental contributions and thanks W. J. Shack and T. F. Kassner for their helpful discussions.

References

1. O. K. Chopra and H. M. Chung, "Aging Degradation of Cast Stainless Steels: Effects on Mechanical Properties," in *Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors*, G. J. Theus and J. R. Weeks, eds., The Metallurgical Society, Warrendale, PA., pp. 737-748 (1988).
2. O. K. Chopra and H. M. Chung, "Effect of Low-Temperature Aging on the Mechanical Properties of Cast Stainless Steels," in *Properties of Stainless Steels in Elevated Temperature Service*, M. Prager, ed., MPC Vol. 26, PVP Vol. 132, American Society of Mechanical Engineers, New York, pp. 79-105 (1988).
3. O. K. Chopra, "Thermal Aging of Cast Stainless Steels: Mechanisms and Predictions," in *Fatigue, Degradation, and Fracture - 1990*, W. H. Bamford, C. Becht, S. Bhandari, J. D. Gilman, L. A. James, and M. Prager, eds., MPC Vol. 30, PVP Vol. 195, American Society of Mechanical Engineers, New York, pp. 193-214 (1990).
4. O. K. Chopra and A. Sather, *Initial Assessment of the Mechanisms and Significance of Low-Temperature Embrittlement of Cast Stainless Steels in LWR Systems*, NUREG/CR-5385, ANL-89/17, Argonne National Laboratory, Argonne, IL (August 1990).
5. A. Trautwein and W. Gysel, "Influence of Long Time Aging of CF-8 and CF-8M Cast Steel at Temperatures Between 300 and 500°C on the Impact Toughness and the Structure Properties," in *Stainless Steel Castings*, V. G. Behal and A. S. Melilli, eds., ASTM STP 756, pp. 165-189 (1982).
6. E. I. Landerman and W. H. Bamford, "Fracture Toughness and Fatigue Characteristics of Centrifugally Cast Type 316 Stainless Steel Pipe after Simulated Thermal Service Conditions," in *Ductility and Toughness Considerations in Elevated Temperature Service*, MPC-8, American Society of Mechanical Engineers, New York, pp. 99-127 (1978).
7. S. Bonnet, J. Bourgoin, J. Champredonde, D. Guttman, and M. Guttman, "Relationship between Evolution of Mechanical Properties of Various Cast Duplex Stainless Steels and Metallurgical and Aging Parameters: An Outline of Current EDF Programmes," *Mater. Sci. and Technol.*, **6**, 221-229 (1990).
8. K. N. Akhurst and P. H. Pumphrey, *The Aging Kinetics of CF3 Cast Stainless Steel in the Temperature Range 300°C to 400°C*, RD/L/3354/R88, Central Electricity Generating Board, Leatherhead, UK (November 1988).
9. P. H. Pumphrey and K. N. Akhurst, "Aging Kinetics of CF3 Cast Stainless Steel in Temperature Range 300 - 400°C," *Mater. Sci. and Technol.*, **6**, 211-219 (1990).

10. G. Slama, P. Petrequin, and T. Mager, "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Castings and Welds," presented at *SMIRT Post-Conference Seminar 6, Assuring Structural Integrity of Steel Reactor Pressure Boundary Components*, August 29-30, 1983, Monterey, CA.
11. Y. Meyzaud, P. Ould, P. Balladon, M. Bethmont, and P. Soulat, "Tearing Resistance of Aged Cast Austenitic Stainless Steel," presented at *Int'l Conf. on Thermal Reactor Safety (NUCSAFE 88)*, October 1988, Avignon, France.
12. P. McConnell and J. W. Sheckherd, *Fracture Toughness Characterization of Thermally Embrittled Cast Duplex Stainless Steel*, Report NP-5439, Electric Power Research Institute, Palo Alto, CA (September 1987).
13. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels in LWR Systems," in *Proc. 18th Water Reactor Safety Information Meeting*, U.S. Nuclear Regulatory Commission, NUREG/CP-0114 Vol. 3, p. 195 (April 1991).
14. O. K. Chopra, *Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems*, NUREG/CR-4513, ANL-90/42, Argonne National Laboratory, Argonne, IL (June 1991).
15. O. K. Chopra, "Studies of Aged Cast Stainless Steel from the Shippingport Reactor," in *Proc. 18th Water Reactor Safety Information Meeting*, U.S. Nuclear Regulatory Commission, NUREG/CP-0114 Vol. 3, 369 (April 1991).
16. H. M. Chung, "Thermal Aging of Decommissioned Reactor Cast Stainless Steel Components and Methodology for Life Prediction," in *Life Assessment and Life Extension of Power Plant Components*, T. V. Narayanan, C. B. Bond, J. Sinnappan, A. E. Meligi, M. Prager, T. R. Mager, J. D. Parker, and K. Means, eds., PVP Vol. 171, American Society of Mechanical Engineers, New York, pp. 111-125 (1989).
17. H. M. Chung and T. R. Leax, "Embrittlement of Laboratory and Reactor Aged CF3, CF8, and CF8M Duplex Stainless Steels," *Mater. Sci. and Tech.* **6**, 249-262 (1990).
18. C. Jansson, "Degradation of Cast Stainless Steel Elbows after 15 Years in Service," presented at *Fontevraud II Intl Symp.*, Sept. 10-14, 1990, Royal Abbey of Fontevraud, France.
19. H. M. Chung and O. K. Chopra, "Long-Term-Aging Embrittlement of Cast Austenitic Stainless Steels - Mechanism and Kinetics," in *Properties of Stainless Steels in Elevated Temperature Service*, M. Prager, ed., MPC Vol. 26, PVP Vol. 132, American Society of Mechanical Engineers, New York, pp. 17-34 (1988).
20. W. L. Server, "Impact Three-Point Bend Testing for Notched and Precracked Specimens," *J. Testing and Eval.* **6**, 29 (1978).
21. L. S. Aubrey, P. F. Wieser, W. J. Pollard, and E. A. Schoefer, "Ferrite Measurement and Control in Cast Duplex Stainless Steel," in *Stainless Steel Castings*, V. G. Behal and A. S. Melilli, eds., ASTM STP 756, pp. 126-164 (1982).

END

DATE
FILMED
314192

I