Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge (MRP-211)

1015013

Final Report, December 2007

EPRI Project Manager
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PRODUCT DESCRIPTION

Irradiation embrittlement (relative to tensile and fracture toughness properties), irradiation-assisted stress corrosion cracking, irradiation-enhanced stress relaxation and creep, and void swelling are potential degradation mechanisms that could affect pressurized water reactor (PWR) internals components. This report describes the current state-of-knowledge, available relevant data, and technical bases for trend model formulations of these mechanisms for long-term functionality evaluations.

Background

The framework for implementing an aging management program for PWR internals component items using inspections and flaw tolerance evaluations to manage degradation issues has been developed and is documented in MRP-134 and MRP-153. The important elements of this framework are the following:

- Screening, categorizing, and ranking PWR internals components for susceptibility and significance to age-related degradation mechanisms
- Performing functionality analyses and safety assessment of PWR internals components to define a safe and cost-effective aging management in-service inspection and evaluation method and strategy

This report describes the trend or lower bound models and the associated technical bases for austenitic stainless steel PWR internals materials for each age-related degradation mechanism considered and used in functionality analyses. Functionality evaluations will be used to refine the categorization and ranking of aged PWR internals components.

Objectives

To assess current knowledge of irradiated material data on age-related degradation mechanisms and to provide state-of-the-art degradation models for functionality analyses of selected PWR internals component items.

Approach

An expert panel was assembled to review relevant degradation data and the associated trend or lower bound models for PWR component items: irradiation embrittlement (relative to tensile and fracture toughness properties), irradiation-assisted stress corrosion cracking, void swelling, and irradiation-enhanced stress relaxation/creep.

Results

The report provides state-of-the-technology data and degradation models for PWR internals austenitic stainless steel materials for each age-related degradation mechanism considered: irradiation embrittlement (relative to tensile and fracture toughness properties), irradiation-assisted stress corrosion cracking, void swelling, and irradiation-enhanced stress relaxation/creep. For each age-related degradation mechanism, an assessment of the data fit to the constitutive models developed in MRP-135 was performed by the expert panel and alternative formulations suggested, as appropriate. Recommended models are presented that provide the trend of degradation with the relevant environmental conditions, such as neutron fluence, neutron flux, temperature, and stress. A number of gaps that still remain in the database were identified for potential future actions.

EPRI Perspective

The EPRI MRP Reactor Internals Focus Group (RI-FG) has been conducting studies to develop technical bases to support aging management of PWR internals, with particular attention to utility license renewal commitments. This report provides models that are recommended to be used in functionality evaluations. These functionality evaluations will be performed to refine the screening of PWR internal components in accordance with *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)* (EPRI report 1008203, June 2005).

Keywords

PWR internals
Aging management
License renewal
Degradation mechanism
Functionality

ABSTRACT

The purpose of this report is to summarize the current state-of-knowledge of neutron irradiation-induced property changes in austenitic stainless steels, principally solution-annealed Type 304 and 304L materials, cold-worked and solution-annealed Type 316 and 316L materials, and Type 308 weld metal. The age-related degradation models were evaluated by an expert panel assembled by EPRI and the Reactor Internals Focus Group (RI-FG). This panel endorsed models to be used in functionality evaluations and suggested modifications of constitutive and trend models in MRP-135. The following age-related degradation mechanisms were addressed in this report:

- Irradiation embrittlement (IE)
- Irradiation-enhanced stress relaxation and creep (SR/IC)
- Void swelling (VS)
- Irradiation-assisted stress corrosion cracking (IASCC)

It has been clearly demonstrated that the tensile properties, which are one of the indicators of irradiation embrittlement, saturate after a neutron exposure of 10 to 20 dpa (~ 6.67 X 10^{21} to 1.33 X 10^{22} n/cm^2, E > 1.0 MeV), and representative models are presented. All fracture toughness data, which is the second indicator of irradiation embrittlement, are bounded by a saturated value for K_{JC} of 38 MPa$\sqrt{m}$ (34.6 ksi $\sqrt{in}$) at neutron exposures greater than 6.67 X 10^{21} n/cm^2 (E > 1.0 MeV), or approximately 10 dpa. Thermal stress relaxation appears to saturate in a short time (<100 hours) with a maximum reduction of 10-20% of the initial bolt preloads at pressurized water reactor (PWR) internals temperatures. Correlations for irradiation creep strain with effective stress times the irradiation dose indicate that a greater creep rate occurs for Type 304 SA material than for Type 316 CW material. A conservative flux-dependent void swelling model has been developed that indicates that significant void swelling may be possible for certain fluence levels and temperatures potentially obtainable in PWR internals, although it is still based solely on extrapolations of fast reactor data. Sufficient PWR internals component data to adequately evaluate this trend are currently lacking.

While tensile properties appear to saturate around 20 dpa (~ 1.33 X 10^{22} n/cm^2, E > 1.0 MeV), laboratory test data indicate that susceptibility to IASCC initiation appears to continue to increase with the irradiation damage level. A lower bound model, developed at the time of the panel meetings, indicated that IASCC crack initiation would not occur in highly irradiated materials loaded to below approximately 50% of yield strength; however, data obtained since the meeting indicate that 50% may not be a bounding value. Appropriate IASCC crack growth data are too scattered, and the expert panel was unable to recommend a suitable model.

A number of gaps in the database were identified for future RI-FG evaluation and consideration.

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- P. Scott (AREVA NP SAS)
- R.E. Gold (Westinghouse)
- J. Rashid (ANATECH)

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LIST OF ACRONYMS, INITIALISMS, AND ABBREVIATIONS

ASTM – American Society for Testing and Materials

B&W – Babcock & Wilcox

BNFL – British Nuclear Fuels Limited

BOR-60– A test reactor at the Research Institute of Atomic Reactors in Dimitrovgrad, Russia

BWR – Boiling Water Reactor

CD – Compact Disc

CE – Combustion Engineering

CEA – French Atomic Energy Commission (Commissariat à l'Energie Atomique)

CERT – Constant Extension Rate Test

CGR – Crack Growth Rate

CIR – Cooperative IASCC Research (Program)

CW – Cold-Worked Condition

DIN – German Institute for Standardization (Deutsches Institut für Normung)

dpa – Displacements Per Atom (a Calculated Measure of Material Damage Due to the Accumulated Fluence)

EBR – Experimental Breeder Reactor

EDF – Electricité de France

EPRI – Electric Power Research Institute

HAZ – Heat-Affected Zone

HWC – Hydrogen Water Chemistry (used for BWR operation)

IASCC – Irradiation-Assisted SCC

IE – Irradiation Embrittlement

IC – Irradiation Creep

IGSCC – Intergranular SCC

JOBB – Joint Owners Baffle Bolt (Program)

ksi – 1000 pound-force per square inch

LEFM – Linear Elastic Fracture Mechanics
LWR – Light Water Reactor
MeV – Million Electron Volts
MHI – Mitsubishi Heavy Industries
MPa – Megapascal
MRP – Materials Reliability Program
Osiris – A Test Reactor at the CEA facilities in Saclay France
Phénix – A Fast Breeder Reactor in France
PNNL – Pacific Northwest National Laboratory
PWR – Pressurized Water Reactor
RI-FG – Reactor Internals Focus Group
SA – Solution-Annealed Condition
SCC – Stress Corrosion Cracking
SFEN – French Nuclear Energy Society (Société Française d'Energie Nucléaire)
SM-2 – A Test Reactor at the Research Institute of Atomic Reactors in Dimitrovgrad, Russia
SR – Stress Relaxation
SSRT – Slow Strain Rate Test
TE – Total Elongation
UE – Uniform Elongation
US – United States
UTS – Ultimate Tensile Strength
VS – Void Swelling
VTT – Technical Research Center of Finland
W – Westinghouse Electric Corporation
YS – Yield Strength

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1 INTRODUCTION

1.1 Report Purpose

The purpose of this report is to summarize the current state-of-knowledge of neutron irradiation-induced property changes in austenitic stainless steels, principally solution-annealed Type 304 and 304L materials, cold-worked and solution-annealed Type 316 and 316L materials, and Type 308 weld metal. This report will provide the basis data and the technical bases for constitutive model formulations of the various age-related degradation mechanisms (e.g., irradiation-assisted stress corrosion cracking, void swelling, irradiation embrittlement[1], etc.) associated with Pressurized Water Reactor (PWR) internals component items. This report was prepared under the direction and sponsorship of the EPRI Materials Reliability Program (MRP) Reactor Internals Focus Group (RI-FG). Valuable input and review comments were received from many sources, in particular from the following RI-FG Expert Panel core members:

- J. Rashid and R. Dunham (ANATECH)
- P. Scott and S. Fyfitch (AREVA)
- R. Gold and M. Burke (Westinghouse)

This report is a key element in an overall strategy for managing the effects of aging in PWR internals using knowledge of internals design, materials and material properties, and applying screening methodologies for the known age-related degradation mechanisms. Related MRP documents include a Framework and Strategy for Managing Aging Effects in PWR Internals [1-1], Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals [1-2] and PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values [1-3].

1.2 Background

The internal component items of PWR vessels that are closest to the fuel (e.g., core barrel, baffle and former plates, bolts, etc.) are subject to high neutron fluxes. Very little data were available for the range of temperatures or neutron flux and fluence levels representative of the end of life of typical PWR internals component items. Conservative calculations estimate temperatures as high as 752°F (400°C), neutron flux at 5.1 X 10^{13} n/cm^{2}s, E > 1.0 MeV, and neutron fluence at

[1] Irradiation Embrittlement is also referred to as Radiation Embrittlement by some authors, but both terminologies refer to the same aging mechanism.

60 years to be 9.6 X 10^{22} n/cm^2, E > 1.0 MeV (~144 dpa).[2] This neutron irradiation changes the microstructure and mechanical properties of these materials such that they harden, lose ductility and toughness, experience irradiation-induced stress relaxation and creep, etc. In addition, these changes seem to be the basis for increased sensitivity to stress corrosion cracking, a phenomenon first observed in boiling water reactors (BWRs) at relatively low fluence levels and denoted as irradiation-assisted stress corrosion cracking (IASCC). Therefore, a variety of testing programs were initiated to provide a comprehensive database for the various age-related degradation mechanisms that affect PWR internals components.

The MRP published a report (MRP-135) [1-4] that developed material constitutive models for irradiated stainless steel, accounting for the effects of plasticity, creep, stress relaxation, void swelling, and embrittlement as functions of temperature, cold-work, and neutron fluence. This effort resulted in a set of material behavior models for use in performing functionality evaluations of PWR internals component items subjected to plant operating conditions.

Until recently, there were few data available for mixed thermal/fast neutron spectra at the high neutron fluence values and temperatures representative of end of life of PWR internals to calibrate the models and assess the predictive ability of the trend models; for example, most of the mechanical property data at high neutron doses were obtained from fast reactor irradiations.

The most recent available data come primarily from two sources:

- Tests performed on PWR irradiated components (bolts, locking bars, baffles, formers, corners, thimble tubes).
- Laboratory tests performed on materials irradiated in experimental light water reactors (Osiris, SM-2) or in experimental fast neutron reactors (EBR II, BOR-60, Phénix).

Table 1-1 contains a listing of experimental reactor operating parameters and the types of test data available. Table 1-2 summarizes the data sources available from operating PWRs.

With this additional data now available from various testing programs, which include the Joint Owners Baffle Bolt (JOBB) program, the Cooperative IASCC Research (CIR) program, and various other MRP testing programs, additional work was initiated in 2006 to interpret the recently completed RI-FG programs and revise or defend the material mechanical behavior and degradation model formulations developed for functionality analyses. Therefore, objectives of this current effort are twofold:

- Critically review and analyze the recently available irradiated material data published in the literature or that was supported by the RI-FG pertinent to PWR internals structures.
- Critically review and reach consensus on modification and/or endorsement of the material constitutive and trend models developed in MRP-135.

[2] The dpa conversion used in this report is based on an estimated conversion factor for PWRs of 10^{22} n/cm^2, E > 1.0 MeV = 15 dpa. Note however that this factor will vary with the flux and location within the PWR internals. [1-5]

A panel of materials experts from the industry was assembled to review the relevant state of knowledge of the material behavior data pertinent to PWR internals and the material constitutive and trend models developed in MRP-135. The results of this work will be used to revise the material constitutive model report (MRP-135, Rev. 1) for use in the functionality evaluations and to aid in establishing the inspection techniques and schedule for the future inspection plan document being prepared by the MRP.

1.3 Report Structure

The data that were used to identify the trends and evaluate the MRP-135 constitutive models for the various age-related degradation mechanisms are provided in the Appendices. Chapter 2 provides the sources and the data trends and identifies gaps in the data where improvement of the database would clarify data trends and improve the accuracy of the predictive models. Chapter 3 provides the constitutive models that are endorsed by the expert panel to be used in functionality analysis. In addition, Chapter 3 also provides a general description of each model along with any associated assumptions and limitations. Chapter 4 provides a summary of the major findings of the expert panel and concluding remarks. The available data are presented in Appendices A through F. Appendix G contains figures of the types of test specimens used to establish the database, Appendix H contains a summary of the available test material chemical compositions, and Appendix I contains a listing of the data references.

Table 1-1
Data Sources for Materials Irradiated in Test Reactors

Reactor	Spectrum	Temperature (°C/°F)	Maximum Dose, dpa (n/cm2, E > 1.0 MeV)	Materials	Data Type
EBR-II	Fast	~ 375/707	~ 30 (~2.00 X 1022)	304SA, 316CW	Tensile, Creep, Fracture Toughness
Phénix	Fast	380/716 to 400/752	~ 30 (~2.00 X 1022)	304SA, 316CW	Tensile, Creep, Fracture Toughness
BOR-60	Fast	~320/608	~ 120 (~8.00 X 1022)	304SA, 316CW, 308L weld, HAZ	Tensile, Creep, Fracture Toughness, Corrosion
SM-2	Mixed (fast + thermal)	~ 300/572	~ 15 (~1.00 X 1022)	304SA, 316CW	Tensile, Corrosion
Osiris	Mixed (fast + thermal)	~ 320608	~ 12 (~8.00 X 1021)	304SA, 316CW	Tensile, Creep, Corrosion

Note: SA = solution-annealed and CW = cold-worked.

Table 1-2
Data Sources for Materials Irradiated in PWRs

Material Component/Source	Material	Laboratory	Temperature (°C/°F)	Dose, dpa (n/cm2, E > 1.0 MeV)
Bolt (Bugey 2)	316LCW	EDF	330/626 to 360/680	10 to 25 (~6.67 X 1021 to 1.67 X 1022)
Bolt (Fessenheim 2)	316LCW	EDF	330/626 to 360/680	12 to 25 (~8.00 X 1021 to 1.67 X 1022)
Bolt (Tihange 1)	316LCW	PNNL EDF	300/572 to 363685	7 to 24 (~4.67 X 1021 to 1.60 X 1022)
Bolt (Farley)	316CW	Westinghouse BNFL	307/584 to 396/745	9 to 19 (~6.00 X 1021 to 1.27 X 1022)
Bolt (Point Beach)	347SA	Westinghouse BNFL	303/577 to 396/745	4 to 15 (~2.67 X 1021 to 1.00 X 1022)
Locking Bar (Fessenheim 2)	304SA	EDF	330/626	25 (~1.67 X 1022)
Locking Bar (Farley)	304SA	Westinghouse BNFL	304/579	20 to 22 (~1.33 X 1022 to 1.47 X 1022)
Locking Clip (Point Beach)	304SA	Westinghouse BNFL	304/579	23 to 26 (~1.53 X 1022 to 1.73 X 1022)
Core Barrel (US PWR Plant)	304SA	Westinghouse	280/536 to 294/561	0.5 (~3.33 X 1020)
Baffle Plate (US PWR Plant)	304SA	Westinghouse	285/545 to 307/584	4 to 17 (~2.67 X 1021 to 1.13 X 1022)
Former Plates (US PWR Plant)	304SA (4 Heats)	Westinghouse	301/574 to 321/610	9 to 13 (~6.00 X 1021 to 8.67 X 1021)
Baffle Corner (Chooz A)	304SA	EDF	~ 300/572	~ 35 (~2.33 X 1022)
Baffle Bolt (Chooz A)	304CW	CEA	~ 300/572	2.5 to 21 (~1.67 X 1021 to 1.40 X 1022)
Thimble Tube (Beaver Valley 1)	316CW	Westinghouse	340/644	12 to 51 (~8.00 X 1021 to 3.40 X 1022)
Thimble Tube (HB Robinson 2)	316CW	Westinghouse	340/644	20 to 30 (~1.33 X 1022 to 2.00 X 1022)
Thimble Tube (Ringhals)	316CW	Westinghouse MHI	340/644, ~ 300/572	0, 35, 70 (0, ~2.33 X 1022, ~4.67 X 1022)
Thimble Tube (Japan)	316CW	MHI	~ 300/572	60 (~4.00 X 1022)

Note: SA = solution-annealed and CW = cold-worked.

1.4 References

[1-1] Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134) - EPRI Report 1008203, 2005.

[1-2] Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153) – EPRI Report 1012082, 2005.

[1-3] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.

[1-4] Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135) – EPRI Report 011127, 2004.

[1-5] Garner, F.A., Greenwood, L.R., and Reid, B.D., "An Assessment of the Possible Role of Transmutation on Development of Irradiation-Assisted Stress Corrosion Cracking in Light Water Reactors," EPRI TR-107159, "Critical Issue Reviews for the Understanding and Evaluation of Irradiation Assisted Stress Corrosion Cracking," Final Report, November 1996

2 PWR INTERNALS AGE-RELATED DEGRADATION MECHANISM DATA SOURCES AND TRENDS

A panel of industry materials experts (chosen by EPRI and the RI-FG) reviewed the available material age-related degradation mechanism data pertinent to the models presented in MRP-135 for evaluation of PWR internals. The review included: irradiation embrittlement data (assessed by the applicable tensile and fracture toughness properties), irradiation-induced stress relaxation and creep data, void swelling data, and irradiation-assisted stress corrosion cracking initiation and growth data. The data review encompassed recently published data identified by the expert panel and the following MRP reports:

- Materials Reliability Program: Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50) – EPRI Report 1000970, 2001.[2.1-1]
- Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.[2.1-2]
- Materials Reliability Program: Characterizations of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), EPRI Report 1003525, 2002.[2.1-3]
- Materials Reliability Program: A Review of Radiation Embrittlement for Stainless Steels for PWRs (MRP-79) - Revision 1 - EPRI Report 1008204, 2004.[2.1-4]
- A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80) – EPRI Report 1003523, 2003.[2.1-5]
- Materials Reliability Program: Stress Corrosion Cracking of High Strength Reactor Vessel Internals Bolting in PWRs (MRP-88) – EPRI Report 1803206, 2003.[2.1-6]
- Materials Reliability Program: A Review of the Cooperative Irradiation Assisted Stress Corrosion Cracking Research Program (MRP-98) – EPRI Report 1002807, 2003.[2.1-7]
- Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples-Material Certification, Fluence, and Temperature (MRP-128) - EPRI Report 1008202, 2004.[2.1-8]
- Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples - Tensile and SSRT Testing (MRP-129) - EPRI Report 1008205, 2004.[2.1-9]
- Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134) - EPRI Report 1008203, 2005.[2.1-10]

- Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135) - EPRI Report 1011127, 2004.[2.1-11]
- Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153) – EPRI Report 1012082, 2005.[2.1-12]
- Materials Reliability Program: Crack Initiation Testing and SSRT Testing of Boris-60 Irradiated Materials and Effect of Hydrogen on IASCC Susceptibility (MRP-159) – EPRI Report 1010096, 2005.[2.1-13]
- Materials Reliability Program: Fracture Toughness Testing of Decommissioned PWR Core Internals Material Samples (MRP-160) – EPRI Report 1012079, 2005.[2.1-14]
- Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.[2.1-15]
- Joint Owners Baffle Bolt Program: JOBB-CD Version 05.12 – EPRI Report 1012083, 2005.[2.1-16]

Several meetings of the expert panel were held to discuss the data validity and the applicability of the data to Westinghouse (W), Combustion Engineering (CE), and Babcock & Wilcox (B&W)-designed PWR internals. The expert panel reviewed the available data that were summarized in a database prepared by AREVA and provided additional data sources for and corrections to the database. In particular, a review and evaluation of the JOBB program final test results was performed. The remaining gaps in the available aging degradation mechanism data were identified by the panel. A summary of data applicable to PWR internals was prepared as a result of the expert panel review and modifications and is presented in the Appendices in tabular form and in the main text in graphical form.

2.1 Tensile Test Data

As noted above, the irradiation embrittlement age-related degradation mechanism is assessed by a review of the applicable tensile and fracture toughness properties. Irradiation embrittlement results from lattice defects caused by neutron bombardment. High-energy neutrons displace atoms from their normal lattice positions and create point defects. Although most point defects (called interstitials and vacancies) are annihilated by recombination, surviving point defects form various irradiated microstructures consisting of dislocations, precipitates, and cavities. Cavities, which are three-dimensional clusters of vacancies, gas atoms (bubbles), or a combination of the two, can be associated with other microstructural features such as precipitates, dislocations, and grain boundaries. These defects and precipitates from irradiation are obstacles to dislocation movement and result in an increased yield and tensile strength and a decreased work-hardening capacity, ductility, and loss of fracture toughness.

Therefore, one indicator of irradiation embrittlement of austenitic stainless steels is associated with the mechanical property changes that are characterized through tensile testing. Tensile properties of irradiated austenitic stainless steels have been widely investigated. However, a 2004 review of available tensile data for irradiated austenitic stainless steels, MRP-79 Revision 1 [2.1-4], indicated that most data were from materials irradiated in various fast reactors. The review also noted that data exceeding fluence levels of 15 dpa (~ 1.0 X 10^{22} n/cm^2, E > 1.0 MeV) in light water reactors (LWRs) were limited. Since that time, additional LWR data for austenitic stainless steels have become available for comparison and evaluation.

The most common austenitic stainless steel alloys in use in PWR internals are solution-annealed Type 304 and cold-worked Type 316. In addition, a number of bolting applications in Westinghouse baffle and former assemblies used cold-worked or solution-annealed Type 347. Table 2.1 presents the composition of Type 304, Type 316, and Type 347 wrought austenitic stainless steels.[2.1-17] These alloys differ only in minor variations in the chromium and nickel concentrations, and by the presence of special alloying additions in Type 316 (molybdenum) and Type 347 (niobium). The chromium and nickel concentration ranges in Type 347 are essentially in the middle of the ranges for Type 304 and Type 316. Molybdenum is added to Type 316 primarily for enhanced corrosion resistance and the niobium addition to Type 347 provides carbide stabilization, thereby giving high resistance to sensitization due to grain boundary carbide precipitation. For the evaluations performed in this report, the expert panel agreed to combine the Type 347 data together with the Type 316 CW data.

Table 2-1
Chemical Composition of Wrought Austenitic Stainless Steels [2.1-17]

Grade	C	Mn	P	S	Si	Ni	Cr	Mo	N	Nb
Type 304	0.07	2.00	0.045	0.030	0.75	8.0-10.5	17.5-19.5	---	0.10	---
Type 316	0.08	2.00	0.045	0.030	0.75	10.0-14.0	16.0-18.0	2.00-3.00	0.10	---
Type 347	0.08	2.00	0.045	0.030	0.75	9.0-13.0	17.0-19.0	---	---	10 x C min., 1.00 max.

2.1.1 Tensile Data Sources

Tensile test data in Appendix A come from the following sources:

- KhN35VT-VD, 14Kh17N2, 08Kh18N10T, Sv08Kh19N10, and Sv04Kh19N11 (Russian stainless steels) specimens irradiated in BOR-60 and tested in an inert environment. (CIR) [2.1-18]
- Type 316, Type 304 and Type 347 stainless steel bolts irradiated in Farley and Point Beach (MRP) [2.1-1]
- Type 316 V-notch specimens in EBR-II (JOBB) [2.1-19]
- Type 316 specimens in Ringhals 2 (MRP) [2.1-3]
- Type 304 and Type 316 specimens irradiated in the SM-2 reactor (JOBB) [2.1-20]
- Type 304 specimens irradiated in Chooz A (JOBB) [2.1-21 and 2.1-22]
- Unirradiated Type 316, Type 316NG, Type 304, Type 347, Type 348, DIN 1.4541, DIN 1.4914, and DIN 1.4981 specimens (MRP) [2.1-23]

- Several variations of Type 304, Type 304H, several variations of Type 316, Type 316L, Type 347, Type 308, Type 321, CF8, X18H10T, N9CW, Nitronic 50, Nitronic 60, Uranus S1N, Alloy 690, Alloy 800, N9, HNI, and NMF18 specimens with various levels of cold work, irradiated in BOR-60 (JOBB) [2.1-24 and 2.1-25]
- Unirradiated and irradiated Type 304, and variations of Type 316 specimens (JOBB) [2.1-26 and 2.1-27]
- Type 304, Type 316 specimens irradiated in EBR-II (JOBB) [2.1-26, 2.1-28, 2.1-29, and 2.1-30]
- Type 347H, Type 316Nb, Type 316, Type 304, Type 316Ti47, N9E, and Type 316SG specimens irradiated in the Osiris reactor (JOBB) [2.1-31]
- Type 304H, Type 316, Type 316Ti47, and Type 316Ti73 specimens irradiated in the Osiris reactor (JOBB) [2.1-32]
- Type 304, Type 304L, Type 316, Type 308L, Type 308, and Type 309 specimens (MRP) [2.1-4]
- Type 304L, Type 316, Type 316L, Type 347, Type 316Ti, Type 304Mo, and Type 304H stainless steel specimens (CIR-MRP) [2.1-7]
- Type 304 stainless steel specimens from baffle and former plates (MRP) [2.1-8 and 2.1-9]
- Type 304, Type 316, Type 316L, Type 347, and Type 308 stainless steel and cast austenitic stainless steel specimens irradiated in BOR-60 (MRP) [2.1-13]

The data fall within the following ranges and parameters:

Materials: Solution-annealed or cold-worked (up to 20%) alloys

Fluence: 0 to 124 dpa (0 to ~8.27 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: room temperature (20°C/68°F) and high temperature (270°C/518°F to 383°C/721°F)

2.1.2 Tensile Test Data Trends

Content deleted – EPRI/MRP Proprietary Information

Content deleted – EPRI/MRP Proprietary Information

Figure 2-1
The Effect of Neutron Fluence on Room Temperature Yield Strength for Solution-Annealed Type 304, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-2
The Effect of Neutron Fluence on Room Temperature Yield Strength for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-3
The Effect of Neutron Fluence on Elevated Temperature Yield Strength for Solution-Annealed Type 304 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-4
The Effect of Neutron Fluence on Elevated Temperature Yield Strength for Solution-Annealed and Cold-Worked Type 316 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-5
The Effect of Neutron Fluence on Room Temperature Ultimate Strength for Solution-Annealed Type 304, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-6
The Effect of Neutron Fluence on Room Temperature Ultimate Strength for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-7
The Effect of Neutron Fluence on Elevated Temperature Ultimate Strength for Solution-Annealed Type 304 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-8
The Effect of Neutron Fluence on Elevated Temperature Ultimate Strength for Solution-Annealed and Cold-Worked Type 316 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-9
The Effect of Neutron Fluence on Room Temperature (20°C/68°F) Uniform Elongation for Solution-Annealed Type 304, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-10
The Effect of Neutron Fluence on Room Temperature (20-25°C/68-77°F) Uniform Elongation for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-11
The Effect of Neutron Fluence on Elevated Temperature (270-280°C/518-716°F) Uniform Elongation for Solution-Annealed Type 304 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-12
The Effect of Neutron Fluence on Elevated Temperature (288-383°C/550-721°F) Uniform Elongation for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-13
The Effect of Neutron Fluence on Room Temperature (20°C/68°F) Total Elongation for Solution-Annealed Type 304, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-14
The Effect of Neutron Fluence on Room Temperature (20-25°C/68-77°F) Total Elongation for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-15
The Effect of Neutron Fluence on Elevated Temperature (270-280°C/518-716°F) Total Elongation for Solution-Annealed Type 304 and Type 347 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-16
The Effect of Neutron Fluence on Elevated Temperature (288-383°C/550-721°F) Total Elongation for Solution-Annealed and Cold-Worked Type 316, Type 347, and Type 348 Stainless Steels

Content deleted – EPRI/MRP Proprietary Information

Figure 2-17
The Effect of Neutron Fluence on Room Temperature Yield Strength for Type 308 Stainless Steel Weld Metal Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-18
The Effect of Neutron Fluence on Elevated Temperature Yield Strength for Type 308 Stainless Steel Weld Metal and Heat-Affected Zone Materials Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-19
The Effect of Neutron Fluence on Room Temperature Ultimate Tensile Strength for Type 308 Stainless Steel Weld Metal Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-20
The Effect of Neutron Fluence on Elevated Temperature Ultimate Tensile Strength for Type 308 Stainless Steel Weld Metal and Heat-Affected Zone Materials Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-21
The Effect of Neutron Fluence on Room Temperature (20-29°C/68-83°F) Total Elongation for Type 308 Stainless Steel Weld Metal Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-22
The Effect of Neutron Fluence on Elevated Temperature (270-330°C/518-626°F) Total Elongation for Type 308 Stainless Steel Weld Metal and Heat-Affected Zone Materials Relative to the MRP Type 304 and Type 316 Models

Content deleted – EPRI/MRP Proprietary Information

Figure 2-23
Comparison of Room Temperature Yield Strength Data Obtained in Fast and Thermal Reactors – Type 304 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-24
Comparison of Elevated Temperature (270-380°C/518-716°F) Yield Strength Data Obtained in Fast and Thermal Reactors – Type 304 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-25
Comparison of Room Temperature Yield Strength Data Obtained in Fast and Thermal Reactors – Type 347/Type 348 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-26
Comparison of Elevated Temperature (288-383°C/550-721°F) Yield Strength Data Obtained in Fast and Thermal Reactors – Type 347/Type 348 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-27
Comparison of Room Temperature Yield Strength Data Obtained in Fast and Thermal Reactors – Type 316 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-28
Comparison of Elevated Temperature (288-383°C/550-721°F) Yield Strength Data Obtained in Fast and Thermal Reactors – Type 316 Stainless Steel

Content deleted – EPRI/MRP Proprietary Information

Figure 2-29
Comparison of Elevated Temperature (330°C/626°F) Yield Strength Data from EDF and B&W Supplied Type 308 Stainless Steel

2.1.3 Tensile Test Data Gaps

Content deleted – EPRI/MRP Proprietary Information

2.1.4 Tensile Test Data References

[2.1-1] Materials Reliability Program: Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50) – EPRI Report 1000970, 2001.

[2.1-2] Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.

[2.1-3] Materials Reliability Program: Characterizations of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), 1003525, 2002.

[2.1-4] Materials Reliability Program: A Review of Radiation Embrittlement for Stainless Steels for PWRs (MRP-79) - Revision 1 - EPRI Report 1008204, 2004.

[2.1-5] A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80) – EPRI Report 1003523, 2003.

[2.1-6] Materials Reliability Program: Stress Corrosion Cracking of High Strength Reactor Vessel Internals Bolting in PWRs (MRP-88) – EPRI Report 1803206, 2003.

[2.1-7] Materials Reliability Program: A Review of the Cooperative Irradiation Assisted Stress Corrosion Cracking Research Program (MRP-98) – EPRI Report 1002807, 2003.

[2.1-8] Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples-Material Certification, Fluence, and Temperature (MRP-128) - EPRI Report 1008202, 2004.

[2.1-9] Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples - Tensile and SSRT Testing (MRP-129) - EPRI Report 1008205, 2004.

[2.1-10] Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134) - EPRI Report 1008203, 2005.

[2.1-11] Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135) - EPRI Report 1011127, 2004.

[2.1-12] Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153) – EPRI Report 1012082, 2005.

[2.1-13] Materials Reliability Program: Crack Initiation Testing and SSRT Testing of Boris-60 Irradiated Materials and Effect of Hydrogen on IASCC Susceptibility (MRP-159) – EPRI Report 1010096, 2005.

[2.1-14] Materials Reliability Program: Fracture Toughness Testing of Decommissioned PWR Core Internals Material Samples (MRP-160) – EPRI Report 1012079, 2005.

[2.1-15] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.

[2.1-16] Joint Owners Baffle Bolt Program: JOBB-CD Version 05.12 – EPRI Report 1012083, 2005.

[2.1-17] A 240/A 240M-07, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications," ASTM Book of Standards, Volume: 01.03, 2007.

[2.1-18] Zamboch, M., Burda, J., Postler, M., Kocik, J., and Keilova, E., "The Program of Lifetime Assessment of VVER Reactor Internals; Testing of Irradiated Materials," (CIR Program NRI in-kind report) Final Report, August 2003.

[2.1-19] DeBuisson, P. and Massoud, J.P., "Task B4 : EBR-II Irradiation Idaho Experiment Tensile Tests and Fractographic Examinations on Notched Specimens, English Summary" Report DMT SEMI/LCMI/RT/99/006/A, Note SRMA 00.118 - May 2002.

[2.1-20] Massoud, J.P. and DeBuisson, P., "Samara Experiments: Results of the Tensile Tests," Rapport EDF-DER HT-41/01/005/A, February 2000.

[2.1-21] Robinot, P., "Cornières de cloisonnement de CHOOZ A. Rapport n°2 : Essais de traction. Essais de ténécité," Rapport EDF D.5716/RBT/RA.95.6341 indice 0, February 1996.

[2.1-22] Dugue, P., "Cornières de cloisonnement de CHOOZ A. Rapport d'étape n°1 : Mesure de l'activité gamma. Mesure de la dureté par méthode Equotip," Rapport EDF D.5716/DGE/RA.95.6100 indice 1, October 1996.

[2.1-23] PWR Materials Reliability Project Analysis of Baffle Former Bolt Cracking in EDF CP0 Plants (MRP-03), TR-112209 Final Report, June 1999.

[2.1-24] Massoud, J.P. and DeBuisson, P., Project T4-98-01 "Internes de cuve REP" - Irradiations dans le réacteur BOR-60. Résultats des essais et examens post-irradiatoires, Rapport EDF HT-27/03/018/A, 2003.

[2.1-25] Massoud, J.P., "Irradiation BORIS 7 in BOR-60 Reactor, Results of Post Irradiation Tests and Investigation," EDF Report HT-27/04/036/A, 2004.

[2.1-26] Pelchat, J., "Caractérisation en traction à l'ambiante et à chaud de sept nuances d'acier austénitique (témoin de l'irradiation IDAHO)," Rapport CEA SRMA 96-1562, 1996.

[2.1-27] LeHong, S. and Massoud, J.P., "Specifications for EDF's Corrosion Tests Program on Irradiated Materials," EDF Report HT-41/00/030/A, 2000.

[2.1-28] Averty, X., Rabouille, O., Coffre, P., et al., "Irradiation IDAHO - Essais de traction et examens fractographiques au MEB sur trois éprouvettes entaillées," Rapport CEA DMT SEMI/LCMI/RT/99/006/A, 1999.

[2.1-29] Bottin, P. and Schill, R., "Expérience IDAHO - document n°2 - Résultats des mesures et essais de traction après irradiation," Rapport CEA DMT 96/058, 1996.

[2.1-30] Coffre, P., David, V., et al., "Expérience IDAHO - document n°3 - Résultats des essais de traction et des observations microscopiques après irradiation," Rapport CEA DMT 96/292, 1996.

[2.1-31] Averty, X., Gennison, M, and Coffre, P., "Expérience Alexandra - Phase 3 (4-6 dpa) - Résultats des essais de traction et mesures de striction à rupture," Rapport CEA DMN SEMI/LCMI/RT/02-032/A, 2002.

[2.1-32] Averty, X., Gennison, M, and Coffre, P., "Expérience Casimir - Phase 4 (10-11 dpa) - Résultats des essais de traction et mesures de striction à rupture," Rapport CEA DMN SEMI/LCMI/RT/02-033/A, 2002.

2.2 Fracture Toughness Data

As explained in Section 2.1, reduction of fracture toughness is the second important consequence of irradiation embrittlement in austenitic stainless steels. When exposed to high-energy neutrons, the fracture toughness of austenitic stainless steel is significantly decreased to a saturation value. Decreasing fracture toughness values lead to decreasing critical crack lengths that are able to be accommodated by the structure without premature failure. A large reduction in fracture toughness of austenitic stainless steels due to neutron irradiation can significantly increase the sensitivity to flaws that are either pre-existing during construction or flaws that might develop during service due to SCC, IASCC, or fatigue.

The fracture toughness of irradiated stainless steels is a function of both temperature and neutron fluence. Irradiation embrittlement of austenitic stainless steels is best characterized through fracture toughness testing, although some Charpy V-notch data are also available. Data at high neutron fluence were generated for materials irradiated in various sources such as test reactors and decommissioned reactor internals component items. This section of the report describes the available data and associated fracture toughness trends applicable to PWR internals materials.

2.2.1 Fracture Toughness Data Sources

Fracture toughness data provided in Appendix B come from the following sources:

- Precracked Type 316 and Type 347 stainless steel bolts irradiated in Farley and Point Beach (MRP) [2.2-1]
- Type 304H and Type 316L round compact tension specimens irradiated at the EBR-II (JOBB) [2.2-2 and 2.2-3]
- Type 304 compact fracture specimens irradiated in Chooz A (JOBB) [2.2-4 and 2.2-5]
- Type 304 compact fracture specimens irradiated in EBR-II (JOBB) [2.2-6]
- Type 304, Type 304L, Type 316, Type 316L, Type 316H, Type 316Ti, Type 308, Type 308L, Type 347, Type 348 specimens, including various CT and bend specimens (MRP) [2.2-6]
- Type 304 specimens from PWR core former and baffle plates and Type 316 flux thimbles from Ringhals (MRP) [2.2-7]
- Type 304 and Type 308 CT specimens (MRP-JOBB) [2.2-8]

<u>The data fall within the following ranges and parameters:</u>

Materials: Solution-annealed and cold-worked alloys, and weld metals

Fluence: 0 to 90 dpa (0 to ~6.00 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: room temperature (20°C/68°F) and elevated temperature (150°C/302°F to 427°C/801°F)

2.2.2 Fracture Toughness Data Trends

Content deleted – EPRI/MRP Proprietary Information

Content deleted – EPRI/MRP Proprietary Information

Figure 2-30
The Effect of Neutron Fluence on Initiation Fracture Toughness –Fast Reactors

Content deleted – EPRI/MRP Proprietary Information

Figure 2-31
The Effect of Neutron Fluence on Initiation Fracture Toughness –Thermal Reactors

Content deleted – EPRI/MRP Proprietary Information

Figure 2-32
The Effect of Neutron Fluence on Fracture Toughness –Fast Reactors

Content deleted – EPRI/MRP Proprietary Information

Figure 2-33
The Effect of Neutron Fluence on Fracture Toughness –Thermal Reactors

2.2.3 Fracture Toughness Data Gaps

Content deleted – EPRI/MRP Proprietary Information

2.2.4 Fracture Toughness Data References

[2.2-1] Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.

[2.2-2] DeBuisson, P. and Massoud, J.P., "Task B4: EBR-II Irradiation - Idaho Experiment - Fracture Toughness Tests on Specimens Irradiated in Idaho Experiment," Rapport DMT SEMI/LCMI/RT/O0/037/A, Note Technique DECM SRMA/OO/2390, November 7, 2001.

[2.2-3] Averty, X. et al., "Task B6:Essais de ténacité sur éprouvettes irradiées CTR5 dans l'expérience IDAHO," Rapport DMT SEMI/LCMI/RT/O0/037/A, Note Technique DECM SRMA/OO/2390, September 16, 2000.

[2.2-4] Robinot, P., "Cornières de Cloisonnement de CHOOZ A. Rapport n°2: Essais de Traction. Essais de Ténécité," Rapport EDF D.5716/RBT/RA.95.6341 Indice 0, February 1996.

[2.2-5] Dugue, P., "Cornières de Cloisonnement de CHOOZ A. Rapport d'étape n°1 : Mesure de l'activité Gamma. Mesure de la Dureté par Méthode Equotip," Rapport EDF D.5716/DGE/RA.95.6100 Indice 1, October 1996.

[2.2-6] Materials Reliability Program: A Review of Radiation Embrittlement for Stainless Steels for PWRs (MRP-79) - Revision 1 - EPRI Report 1008204, 2004.

[2.2-7] Materials Reliability Program: Fracture Toughness Testing of Decommissioned PWR Core Internals Material Samples (MRP-160) – EPRI Report 1012079, 2005.

[2.2-8] Tang, H.T., "PWR Internals Testing Status Review," MRP RI-FG Meeting, San Francisco, CA, July 10, 2007.

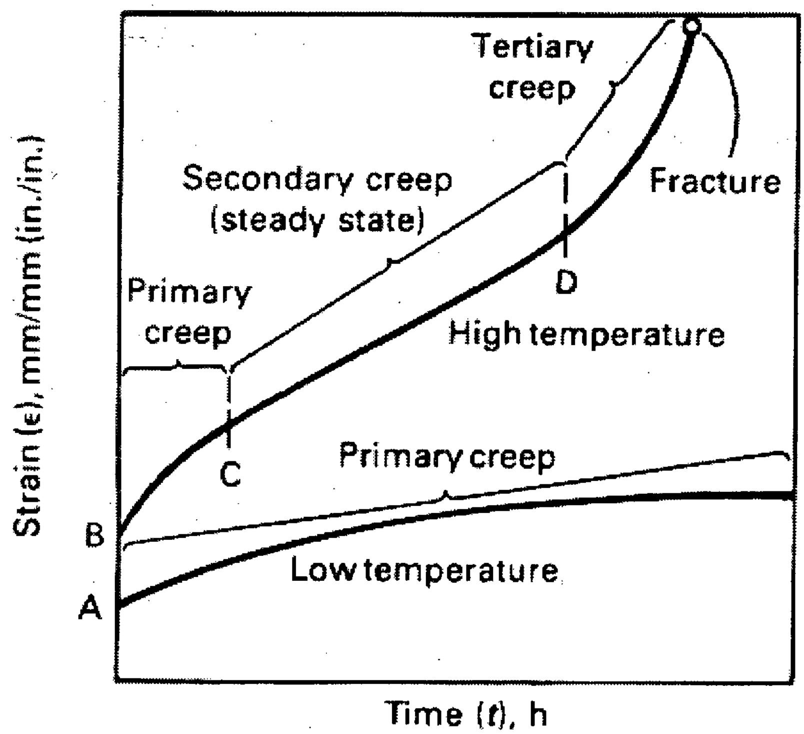
[2.2-9] Hamilton, M., Huang, F-H, Yang, W.J.S., and Garner, F.A., "Mechanical Properties and Fracture Behavior of 20% Cold-Worked 316 Stainless steel Irradiate to Very High Neutron Exposures," Influence of Radiation in Material Properties: 13th International Symposium (Part II), ASTM STP 956, ASTM, Philadelphia, 1987.

2.3 Thermal and Irradiation Creep/Stress Relaxation Data

The general mechanisms of stress relaxation and creep are discussed in MRP-175.[2.3-1] The MRP sponsored a project in 2001 to evaluate the available data on stress relaxation and creep. The effort is compiled in MRP-50.[2.3-2]

For PWR internals, concern over thermal stress relaxation (primary creep, see Figure 2-34) is associated with bolted joints and coil or leaf springs potentially leading to excessive wear or fatigue failure. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. Only a few thermal stress relaxation tests by primary creep have been reported in the literature for austenitic stainless steels at PWR internals temperatures (approximately 288°C/550°F to 343°C/650°F).

Irradiation-enhanced creep (or, more simply irradiation creep) is an athermal process that depends on the neutron fluence and stress and can also be affected by void swelling should it occur. It manifests itself in two roles that impact PWR component functionality. In the first case, irradiation creep operates to mitigate potential increases in stress caused by void swelling; especially problems that might be caused by gradients in swelling within a given component item. In the second case, however, it is important to note that irradiation creep exists even in the absence of void swelling and it can have potential consequences arising primarily from relaxation of preloaded components (excessive wear or fatigue) or from sustained pressure differences across a component such as a plate (fluid flow or heat transfer). A large amount of fast and thermal test reactor stainless steel material test data exist for irradiation creep and empirical equations for the steady-state creep are well developed. The RI-FG database consists of data for a variety of austenitic stainless steel alloys, including Types 304 SA and 316 CW.



(A and B denote the elastic strain on loading; C denotes transition from primary-to-secondary creep; D denotes transition from secondary-to-tertiary creep)

Figure 2-34
Schematic Representation of Creep Curves under Constant Load

2.3.1 Thermal and Irradiation Creep/Stress Relaxation Data Sources

Irradiation creep/stress relaxation data provided in Appendix C come from the following sources:

- 1Kh18N9, DIN 17006, Type 304, 1Kh18N9T stainless steel (MRP) [2.3-2]
- Type 304H, Type 316E, Type 316TiE stainless steel (Casimir) [2.3-3]
- Type 304H, Type 316E, Type 316TiE stainless steel (Casimir) [2.3-4]
- Type 304L, Type 316, Type 316Ti, Type 316 HPSi, Type 316 HP stainless steel [2.3-5]

<u>The data for Type 304 and Type 316 materials fall within the following ranges and parameters:</u>

Materials: Solution-annealed and cold-worked alloys

Fluence: 0 to ~120 dpa (0 to ~8.00 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: 24°C/75°F to 482°C/900°F

2.3.2 Thermal and Irradiation Creep/Stress Relaxation Data Trends

Content deleted – EPRI/MRP Proprietary Information.

2.3.3 Thermal and Irradiation Creep/Stress Relaxation Gaps

Content deleted – EPRI/MRP Proprietary Information.

2.3.4 Thermal and Irradiation Creep/Stress Relaxation References

[2.3-1] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.

[2.3-2] Materials Reliability Program: Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50) – EPRI Report 1000970, 2001.

[2.3-3] Averty, X., Pinte, G., et al., "Experience CASIMIR - Métrologie et Pesée Après la Phase 4 (10 - 11 dpa)," Rapport CEA SEMI/LCMI/RT/02-009/A, 2002.

[2.3-4] Averty, X., Pinte, G., et al., "Experience CASIMIR - Métrologie et Pesée Après la Phase 3 (7 - 9 dpa)," Rapport CEA SEMI/LCMI/RT/01-019/A, 2001.

[2.3-5] Massoud, J.P. and DeBuisson, P., "Project T4-98-01 "Internes de Cuve REP" - Irradiations dans le Réacteur BOR-60, Résultats des Essais et Examens Post-Irradiatoires," Rapport EDF HT-27/03/018/A, 2003.

Content deleted – EPRI/MRP Proprietary Information

Figure 2-35
Irradiation Creep/Stress Relaxation Data and Model Fits

2.4 Void Swelling Data

Irradiation-induced void swelling, hereafter call void swelling (VS), occurs under some conditions in all structural alloys used in various reactor types, but is especially prevalent in austenitic stainless steels and nickel-base alloys. Void swelling is a potential concern for PWRs operating at elevated temperatures in that it produces volumetric and thus dimensional changes, and when acting in concert with irradiation creep, it may produce distortion of structural component items. In addition, if swelling exceeds ~5 to 10%, an additional concern begins to develop with respect to fracture toughness. Whereas the irradiation-induced changes in mechanical properties typically saturate at relatively low neutron exposures, a new form of embrittlement associated with void swelling arises with increasing swelling.

Swelling in austenitic stainless steels is caused by the creation of small cavities that appear under certain conditions of temperature, dose rate, and dose. Swelling has typically been a concern for fast reactors because of high doses and operating temperatures, but a small amount (up to 0.24%) of void swelling has also been observed under PWR conditions, primarily in highly irradiated baffle-to-former bolts. Estimates of the level of void swelling under PWR conditions have been based on extrapolations of data from fast reactor irradiation, albeit from reflector positions where neutron flux values (between 4 x 10^{-4} and 10^{-7} dpa/s) and temperatures (370-430°C/698-806°F) were, nevertheless, still much higher than PWR conditions. Void swelling measurements of PWR internals component items to date (primarily baffle bolts) although small are consistent with extrapolations based on the currently available predictive model. This section of the report describes the available void swelling data and trends considered applicable to PWR internals materials.

2.4.1 Void Swelling Data Sources

Void swelling data provided in Appendix D come from the following sources:

- Type 316 stainless steel bolts irradiated in Ringhals 2 (MRP) [2.4-1]
- Type 304, Type 316, and Type 347 stainless steel bolts irradiated in Farley, Point Beach, Tihange 1, and Chooz A (MRP) [2.4-2 through 2.4-5]
- Type 304, Type 316, and Type 316Ti stainless steel bolts irradiated in EBR-II (MRP) [2.4-6 through 2.4-8]
- Type 304, Type 316, and Type 316Ti stainless steel irradiated in Alexandra and Boris (MRP) [2.4-8 and 2.4.9]
- Type 304, Type 316, Type 316L, and Type 347 Stainless Steel (JOBB) [2.4-8]

<u>The data fall within the following ranges and parameters:</u>

Materials: Cold-worked and solution-annealed

Fluence: 0 to 76 dpa (0 to ~5.07 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: 290°C/554°F to 376°C/709°F

2.4.2 Void Swelling Data Trends

Content deleted – EPRI/MRP Proprietary Information.

Table 2-2
Thermal Reactor Void Swelling Data

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Table 2-2 (continued)
Thermal Reactor Void Swelling Data

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2.4.3 Void Swelling Data Gaps

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Figure 2-36
Void Swelling Data for Austenitic Stainless Steels Irradiated in Thermal Reactors (Mainly PWRs)

2.4.4 Void Swelling Data References

[2.4-1] Materials Reliability Program: Characterizations of Type 316 Cold-Worked Stainless Steel Highly Irradiated Under PWR Operating Conditions (MRP-73), EPRI Report 1003525, 2002.

[2.4-2] Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.

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[2.4-4] Garner, F., Edwards, D.J., et al., "Recent Developments Concerning Potential Void Swelling of PWR Internals Constructed from Austenitic Stainless Steels," Proceedings of the International Symposium Fontevraud V: Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors, SFEN (French Nuclear Energy Society), Paris, France, September 2002.

[2.4-5] Goltrant, O., "Résultats et Synthèse des Essais de Corrosion Effectués sur des Eprouvettes Prélevées dans les vis de Cloisonnement de Bugey 2," Note Technique EDF-GDL D.5716/GRT/BSI/RA 95.6286, September 1995.

[2.4-6] DeBuisson, P. and Massoud, J.P., "Task B4 : EBR-II Irradiation - Idaho Experiment - Fracture Toughness Tests on Specimens Irradiated in Idaho Experiment," Rapport DMT SEMI/LCMI/RT/O0/037/A, Note Technique DECM SRMA/OO/2390, November 7, 200l.

[2.4-7] Averty, X. et al., "Task B6:Essais de Ténacité sur Eprouvettes Irradiées CTR5 dans l'expérience IDAHO," Rapport DMT SEMI/LCMI/RT/O0/037/A, Note Technique DECM SRMA/OO/2390, September 16, 2000.

[2.4-8] Massoud, J.P., "Internes REP : Synthèse des propriétés mécaniques sous irradiation," Rapport EDF HT-27/05/028/A, 2005.

[2.4-9] Pokor, C., DeBuisson, P. and Massoud, J.P., "Task B5: Status of the Characterization of the Irradiated Materials," Note SRMA 01.0116 + Meeting EPRI JoBB / EDF, May 2002.

2.5 IASCC Initiation Data

Irradiation-assisted stress corrosion cracking (IASCC) is a mechanism for which materials exposed to neutron irradiation become more susceptible to SCC with increasing fluence. Despite numerous investigations and research effort, details of the IASCC mechanism in PWR internals remain poorly understood. The current consensus is that IASCC results from the synergistic effects of irradiation damage to the material, the aggressive water environment, and the stress state. The evidence is that radiolysis effects do not play any role in hydrogenated PWR primary water. At present, interactions between these variables have not been adequately quantified and a primary IASCC controlling mechanism has not been identified. A screening neutron fluence value of $\geq 2 \times 10^{21}$ n/cm^2 (E > 1.0 MeV) or ~ 3 dpa has been identified for initiation of IASCC in austenitic stainless steels in MRP-175, based on relatively limited data in PWR internals.

In the 1990s, it was recognized that there was a paucity of IASCC data. This provided the impetus to develop the data to characterize IASCC behavior of PWR internals materials with respect to neutron fluence, temperature, degree of cold-work, and material type. Experimental programs were conducted through the CIR, the MRP, and the JOBB programs. These programs have been completed or are nearly complete. The results are summarized in this section of the report.

2.5.1 IASCC Initiation Data Sources

- IASCC initiation data provided in Appendix E come from the following sources:
- Type 304, Type 347, and Type 316 SSRT specimens irradiated at Farley and Point Beach, tested in a PWR environment (MRP) [2.5-1]
- Type 316 SSRT and O-ring specimens irradiated at Ringhals 2 tested in PWR and inert environments, (MRP) [2.5-2]
- Type 316 tensile specimens irradiated at Bugey 2 tested in PWR and inert environments (MRP) [2.5-3]
- Type 304 tensile specimens irradiated at Chooz A and tested in PWR and inert environments (JOBB) [2.5-4, 2.5-5, 2.5-6, and 2.5-7]
- Type 316 SSRT and O-ring specimens irradiated at BOR 60 and tested in a PWR environment (MRP) [2.5-8]
- Proton irradiated Type 304 CERT specimens tested in PWR and BWR normal water chemistry (2 ppm O_2) environments (CIR) [2.5-9]
- Type 304 SSRT specimens tested in PWR and inert environments, and water + various oxygen and hydrogen concentrations (CIR) [2.5-10 and 2.5-11]
- Type 304 SSRT specimens tested in an inert environment and water + 200 ppb oxygen (CIR) [2.5-12]
- KhN35V, 14Kh17N2, 08Kh18N10, Sv08Kh19N10, Sv04Kh19N11 SSRT specimens irradiated in BOR-60 and tested in a PWR environment (CIR) [2.5-13]

- Type 316, Type 304, and Type 347 SSRT and O-ring specimens irradiated at BOR-60 and tested in a PWR environment (MRP-CIR) [2.5-14]
- Type 316, Type 304, Type 308 SSRT specimens irradiated at BOR-60 and tested in a PWR environment (JOBB) [2.5-15]
- Type 304 and Type 308 SSRT specimens irradiated at BOR-60 and tested in a PWR environment (JOBB) [2.5-16]
- Type 304 CERT specimens tested in a BWR HWC environment (MRP) [2.5-17]
- Type 304 SSRT specimens taken from a PWR core barrel, baffle and formers (MRP) [2.5-18]
- Type 304, Type 308, Type 316, Type 347 SSRT and O-ring specimens irradiated at BOR-60 and tested in a PWR environment (MRP) [2.5-19 and 2.5-20]
- Type 316 SSRT and O-ring specimens taken from PWR thimble tubes (MRP) [2.5-21]

The data fall within the following ranges and parameters:

Materials: Cold-worked and solution-annealed alloys, HAZ, and weld metals

Fluence: 0 to 65 dpa (0 to ~ 4.33 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: elevated temperature (290°C/554°F to 340°C/644°F)

Time: 0 to 4000 hours

2.5.2 IASCC Initiation Data Trends

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Content deleted – EPRI/MRP Proprietary Information

Figure 2-37
IASCC Flaw Initiation Data Stress Versus Time to Cracking – Constant Load Tests

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Figure 2-38
IASCC Flaw Initiation Stress Versus Dose – Constant Load Tests

2.5.3 IASCC Initiation Data Gaps

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2.5.4 IASCC Initiation References

[2.5-1] Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.

[2.5-2] Materials Reliability Program: Characterizations of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), EPRI Report 1003525, 2002.

[2.5-3] Goltrant, O., "Résultats et Synthèse des Essais de Corrosion Effectués sur des Eprouvettes Prélevées dans les vis de Cloisonnement de Bugey 2," Note Technique EDF-GDL D.5716/GRT/BSI/RA 95.6286, September 25, 1995.

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[2.5-13] Zamboch, M. et al., "The Program of Lifetime Assessment of VVER Reactor Internals; Testing of Irradiated Materials," Final Report, August, 2003.

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[2.5-15] Massoud, J.P. and Shogan, R., "Study of the Irradiation Assisted Stress Corrosion Cracking (IASCC) Behavior for Highly Irradiated Austenitic Stainless Steels : $5x10^{-8}s^{-1}$, $1x10\text{-}7s^{-1}$, and $2.5x10^{-7}$ s^{-1}at 340°C and $5x10^{-8}$ s^{-1}at 360°C SSRT Test Results Specimens B38, B36, B37, B7X, FD3, FE3," Contract N° T41/E57141 Interim Report, 2004.

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2.6 IASCC Growth Data

Subcritical crack growth by IASCC is affected by stress intensity, neutron fluence level, and temperature. Experimental programs were conducted through the CIR, the MRP, and the JOBB programs that generated crack growth data for various stainless steels; however, the data are still quite sparse. The available results are summarized in this section of the report.

2.6.1 IASCC Growth Rate Data Sources

IASCC growth data provided in Appendix F come from the following sources:

- Type 347, and Type 316 precracked bolt specimens irradiated at Farley and Point Beach, tested in a PWR environment (MRP) [2.6-1]
- Type 304 0.5T CT specimens irradiated in BWRs and tested in a PWR environment (CIR) [2.6-2 and 2.6-3]
- Type 304L 0.5T CT specimens irradiated in BWRs and tested in water + 2000 ppb oxygen or in water + 1580 to 3160 ppb hydrogen (CIR) [2.6-4]
- Type 304L and Type 316L 0.5T and 1.0T CT specimens tested in a PWR environment (CIR) [2.6-5]
- Type 304 CT specimens irradiated in Chooz A and tested in a PWR environment (CIR) [2.6-6 and 2.6-7]
- Type 304, Type 316L, Type 347 0.5T CT specimens with various levels of cold work and tested in a PWR environment (CIR) [2.6-8 and 2.6-9]
- Type 304 CT specimens irradiated in Halden and tested in a high purity water (CIR) [2.6-10]
- Type 304L, Type 316L, Type 348, Type 304 CT specimens tested in PWR, BWR, inert, dissolved 40 ppm oxygen and dissolved 3.2 ppm hydrogen environments (MRP) [2.6-11]

<u>The data fall within the following ranges and parameters:</u>

Material: Cold-worked and solution-annealed alloys

Fluence: 0 to 32.9 dpa (0 to ~2.19 X 10^{22} n/cm^2, E > 1.0 MeV)

Temperature: 280°C (536°F) to 340°C (644°F)

2.6.2 IASCC Growth Rate Data Trends

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2.6.3 IASCC Growth Rate Data Gaps

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Figure 2-39
The Effect of Stress Intensity on IASCC Crack Growth Rate

2.6.4 IASCC Growth Rate References

[2.6-1] Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51) – EPRI Report 1003069, 2001.

[2.6-2] Davis, R.B., "Evaluation of Fundamental Linkage Among SCC Phenomena," Cooperative IASCC Research (CIR) Program Draft Final Report, EPRI 4068-21, May 2002.

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[2.6-6] Karlsen, T.M., Jenssen, H., and Espeland, M., "Development of In-core Constant Load Testing Procedure for Evaluation of IASCC," Halden, October 30, 2002.

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[2.6-8] Castaño, M.L., García, M.S., de Diego, G., and Gómez-Briceño, D., "Hardening Effect on IGSCC of Austenitic Stainless Steels: Implications to the IASCC," CIR II Steering Committee Meeting, May 15, 2002, Att.26.

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[2.6-10] Gruber, G., Chopra, O., and Shack, W., "Crack Growth Tests on Austenitic Stainless Steels Irradiated in the Halden Reactor," CIR II Steering Committee Meeting, May 15, 2002, Att.18.

[2.6-11] Materials Reliability Program: A Review of the Cooperative Irradiation Assisted Stress Corrosion Cracking Research Program (MRP-98) – EPRI Report 1002807, 2003.

3 MATERIAL CONSTITUTIVE MODELS FOR PWR INTERNALS FUNCTIONALITY ANALYSES

The expert panel reviewed the data described in Chapter 2 (also see Appendices A through F), discussed how well these data fit the model formulations developed in MRP-135, and provided suggestions and alternative approaches to improve the models. Alternative formulations for the modeling of initial (unirradiated) mechanical properties, irradiated mechanical properties, IASCC initiation, irradiation-assisted stress relaxation/creep, and void swelling were considered. The models endorsed by the expert panel are discussed in this chapter of the report. The endorsed models are anticipated to be included in a future revision of MRP-135.

The objective of developing constitutive models is to provide irradiated material-specific age-related degradation mechanism models for use in finite-element-based analytical methods for functionality evaluations of particular internals component items that are subjected to specific neutron fluence, temperature and stress ranges. Functionality evaluations of PWR internals component items will require characterizations of irradiated mechanical properties and fracture toughness, void swelling, irradiation-induced stress relaxation and creep, and IASCC to be performed. These evaluations can then be used to assess the adequacy of present degradation management methods or to prioritize component items for enhanced or augmented inspections.

Trend models were developed for the above age-related degradation mechanisms based on mechanistic considerations and were calibrated using the data in Appendices A through F. All material properties were characterized as a function of neutron fluence and temperature for the various stainless steel alloys to the extent possible as dictated by the breadth of the database. In general, the models provide either reasonable trends or lower bounds of the available data. Fracture toughness and irradiation-induced stress relaxation/creep data are characterized using a lower bound approach as are estimates of crack initiation stress thresholds used for IASCC. The present void swelling model appears to conservatively overestimate the known swelling data.

The exceptions and limitations associated with the age-related degradation models are also discussed in this chapter of the report. In general, the precision of the models are limited by the robustness and breadth of the database.

The following subsections present details of the models proposed for functionality evaluations for the following age-related degradation mechanisms:

- Irradiation Embrittlement, which is represented by the combined modeling of the tensile properties and fracture toughness (Sections 3.1 and 3.2)
- Irradiation-Enhanced Stress Relaxation and Creep (Section 3.3)

- Void Swelling (Section 3.4)
- Irradiation-Assisted Stress Corrosion Cracking (Section 3.5)

3.1 Tensile Data Models

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Table 3-1
Tensile Property Coefficients for the EDF Trend Model at 330°C (626°F)

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3.2 Fracture Toughness Model

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3.3 Irradiation-Enhanced Stress Relaxation and Creep Model

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Figure 3-1
Linear Regression Fits of Available Irradiation Creep Data

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Figure 3-2
Type 304 SA & Type 316 CW Irradiation-Induced Creep Behavior at 330°C (626°F) and 100 MPa (14.5 ksi)

3.4 Void Swelling Model

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Table 3-2
Estimated Void Swelling Predictions for Solution-Annealed Type 304 Material

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3.5 IASCC Initiation Model

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Figure 3-3
IASCC Yield Stress Multiplication Factor, S(d), as a Function of Dose

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Figure 3-4
IASCC Susceptibility Stress for Type 304 SA at 330°C (626°F) as a Function of Dose

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Figure 3-5
IASCC Susceptibility Stress for Type 316 CW at 330°C (626°F) as a Function of Dose

Comments Regarding an IASCC Growth Model

Content deleted – EPRI/MRP Proprietary Information

3.6 References

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4 SUMMARY

A panel of materials experts was assembled by the MRP RI-FG to review and evaluate the most recent data for age-related degradation evaluations of PWR internals. This report provides the state-of-the-technology data and degradation trend or lower bound models for austenitic stainless steel PWR internals materials for each age-related degradation mechanism considered: irradiation embrittlement (relative to tensile and fracture toughness properties), irradiation-enhanced stress relaxation and creep, void swelling, and irradiation-assisted stress corrosion cracking. A complete database of these properties was prepared and is provided in Appendices A through F. The available data come primarily from two sources:

- Tests performed on irradiated PWR component items removed from service (bolts, locking bars, baffle, formers, corners, and thimble tubes).
- Laboratory tests performed on materials irradiated in experimental light water reactors (Osiris, SM-2) or in experimental fast neutron reactors (EBR II, BOR-60, Phenix).

For each age-related degradation mechanism, an assessment of the data fit to the constitutive models developed in MRP-135 was performed by the expert panel and alternative or revised formulations suggested, as appropriate. Recommended models are presented that provide the trend of degradation with the relevant environmental conditions such as neutron fluence, neutron flux, temperature, and stress.

It has been clearly demonstrated that the tensile properties saturate after a neutron exposure of 10 to 20 dpa (~ 6.67 X 10^{21} to 1.33 X 10^{22} n/cm^2, E > 1.0 MeV); the representative models recognize this feature. All fracture toughness data are bounded by a saturated value for K_{JC} of 38 MPa$\sqrt{m}$ (34.6 ksi$\sqrt{in}$) at neutron exposures greater than 6.67 X 10^{21} n/cm^2 (E > 1.0 MeV) or approximately 10 dpa. Thermal stress relaxation due to primary creep appears to saturate in a short time (<100 hours) with a maximum reduction of 10-20% of the initial bolt preloads at PWR internals temperatures. Correlations for irradiation creep strain with effective stress times the irradiation dose indicate that the irradiation-enhanced creep rate is slightly greater for Type 304 SA than for Type 316 CW material. The irradiation creep correlations are consistent with other literature data, although the slopes are ~ 14% higher than previously observed for fast reactor data alone. However, void swelling has definitely been shown not to be involved in enhancing irradiation creep in the current set of data.

A conservative flux-dependent void swelling model has been developed, which indicates that measurable void swelling may be possible for certain fluence levels and temperatures potentially attainable in PWR internals; however, the model is still based solely on extrapolations of data from a fast reactor reflector. Sufficient PWR internals component item data to adequately evaluate this trend are currently lacking.

While the tensile properties appear to saturate around 20 dpa (~1.33 X 10^{22} n/cm^2, E > 1.0 MeV), laboratory test data indicate that susceptibility to initiation of IASCC appears to continue to increase with the irradiation damage level. A lower bound model, using the available data at the time of the expert panel meetings, was developed for functionality assessments that indicated IASCC crack initiation would not occur in highly irradiated materials loaded to below approximately 50% of the as-irradiated yield strength. Additional data obtained since that time indicate that 50% may not be a lower bounding value. In addition, sufficient IASCC crack growth data are lacking and the expert panel was unable to recommend a suitable model.

A number of gaps, which are noted in Chapter 2, were identified in the available database, especially with respect to a full understanding of the age-related degradation expected at high damage levels (from 100 to 150 dpa or ~6.67 X 10^{22} to 1.00 X 10^{23} n/cm^2, E > 1.0 MeV) and in the case of void swelling, the appropriate temperature and dose rate ranges. However, despite the data gaps, it was the conclusion of the expert panel that the proposed models permit the functionality assessment of PWR internals to be performed in order to identify the aging degradation of component items of most concern with reasonable confidence.

A
APPENDIX A: TENSILE PROPERTY DATA

Table A-1
Tensile Test Data

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B
APPENDIX B: FRACTURE TOUGHNESS DATA

Table B-1
Fracture Toughness Testing - Fast Reactors

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Table B-2
Fracture Toughness Testing - Thermal Reactors

Content deleted – EPRI/MRP Proprietary Information

C
APPENDIX C: IRRADIATION CREEP DATA

Table C-1
Irradiation Creep Data

Content deleted – EPRI/MRP Proprietary Information

D
APPENDIX D: VOID SWELLING DATA

Table D-1
Void Swelling Data

Content deleted – EPRI/MRP Proprietary Information

E

APPENDIX E: IASCC DATA

Table E-1
IASCC Initiation Tests: O-Ring and Constant Load Tests

Content deleted – EPRI/MRP Proprietary Information

Table E-2
IASCC Slow Strain Rate Tests

Content deleted – EPRI/MRP Proprietary Information

F
APPENDIX F: IASCC GROWTH DATA

Table F-1
IASCC Growth Rates: Crack Growth Testing

Content deleted – EPRI/MRP Proprietary Information

G

APPENDIX G: TEST SPECIMEN DESIGNS

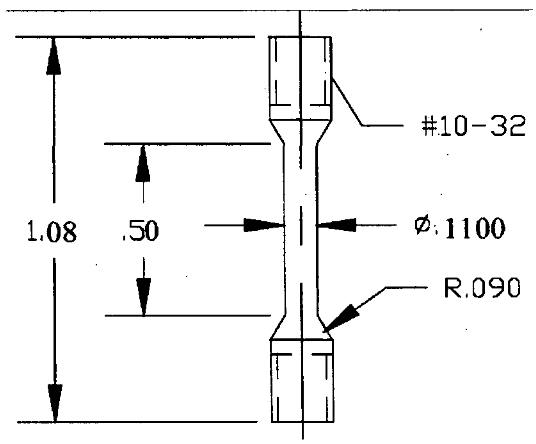

Figure G-1
Farley and Point Beach Baffle-Former Bolt Tensile and SSRT Specimen Design [1]

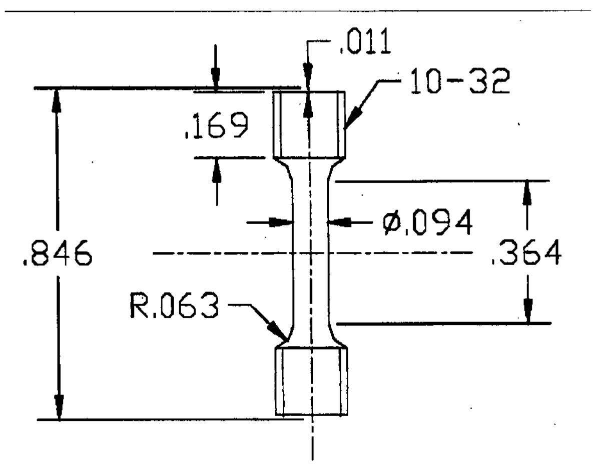

Figure G-2
Farley Tensile and SSRT Specimen Design [1]

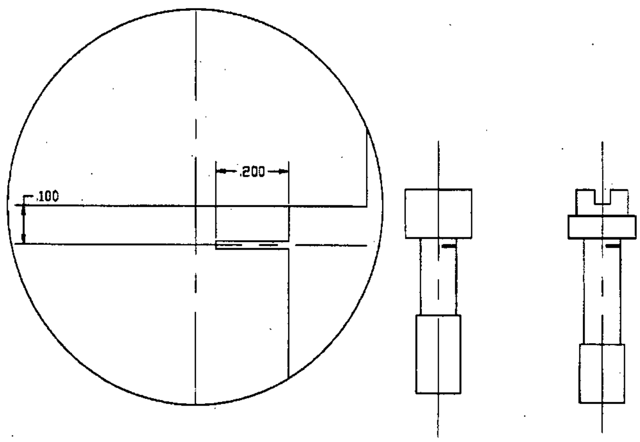

Figure G-3
Farley and Point Beach Specimen Notch Configuration/Dimension [1]

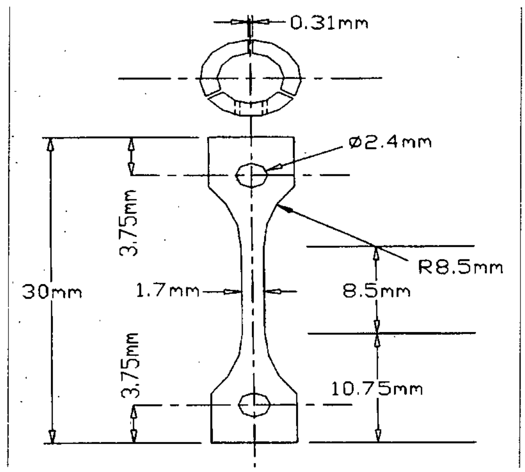

Figure G-4
Ringhals 2 Tensile and SSRT Specimen Design [2]

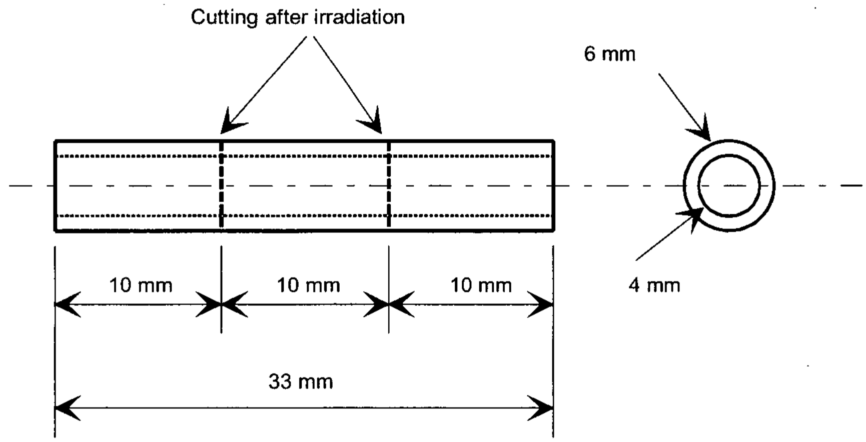

Figure G-5
Tube for Corrosion Tests (Constant Strain Tests) [2, 9, 26]

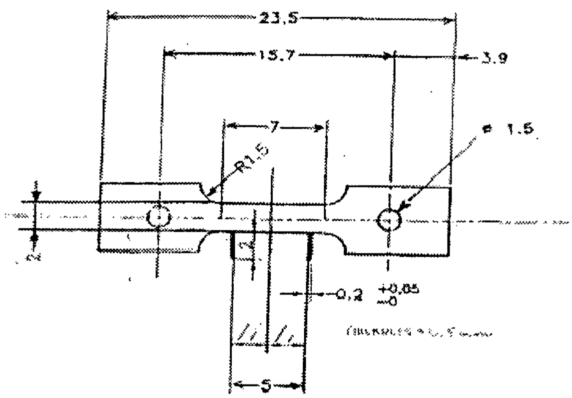

Figure G-6
Bugey 2 Baffle Bolt Head Test Specimen Design [6]

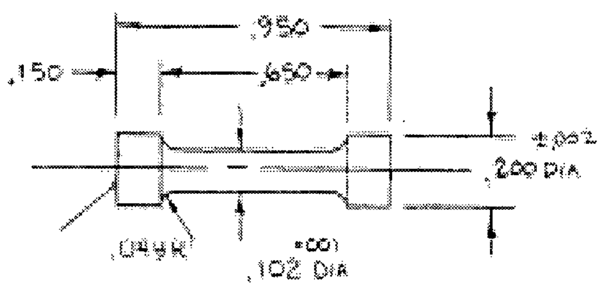

Figure G-7
Bugey 2 – Slow Strain Rate Tensile Specimen Configuration [6]

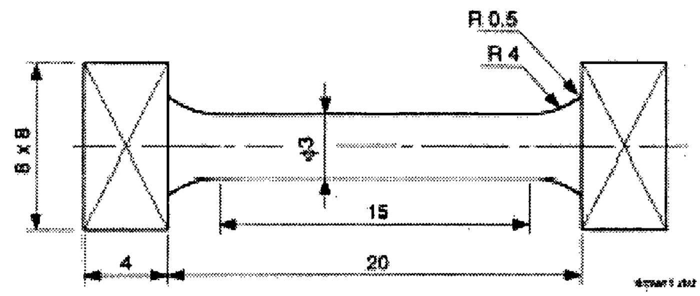

Figure G-8
Chooz A Constant Load Test Specimen [8]

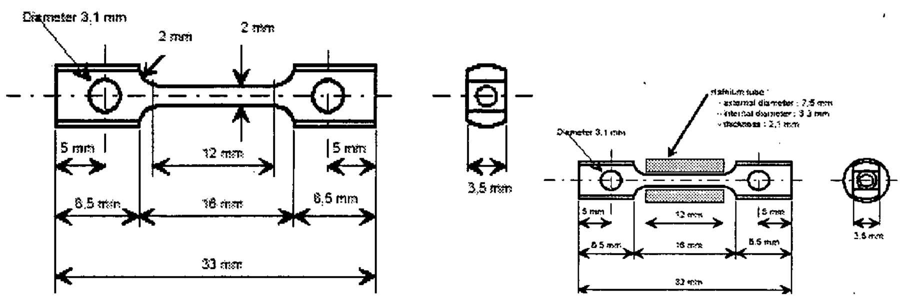

Figure G-9
Samara Tensile Specimen without Hafnium Shielding [9]

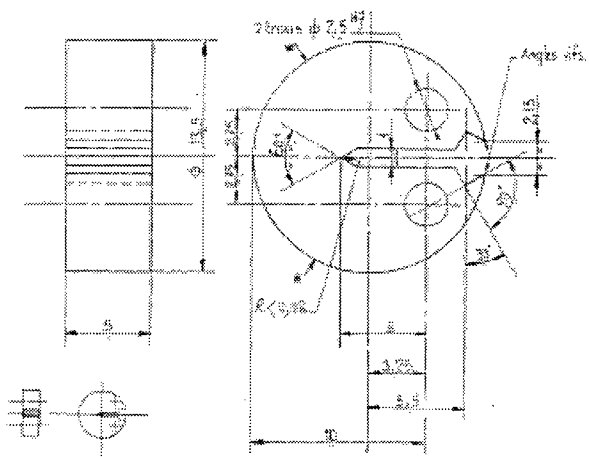

Figure G-10
CTR5 Specimen [12]

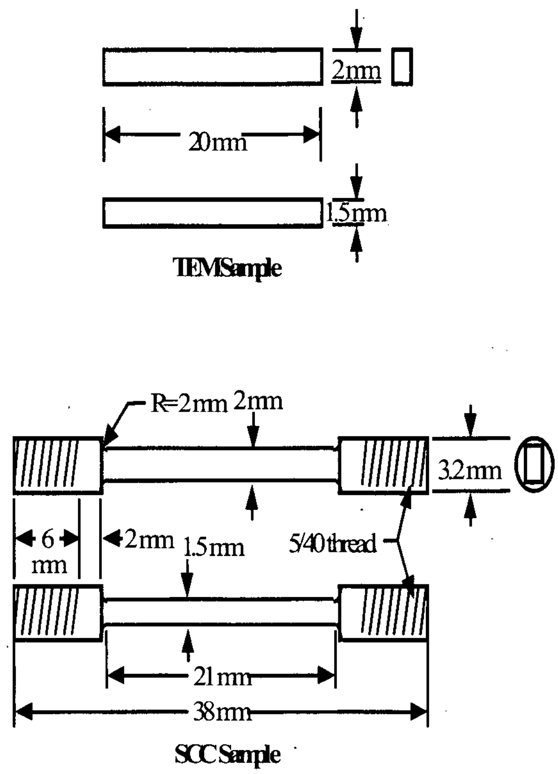

Figure G-11
Schematic of Samples for Proton Irradiations (CERT) [15]

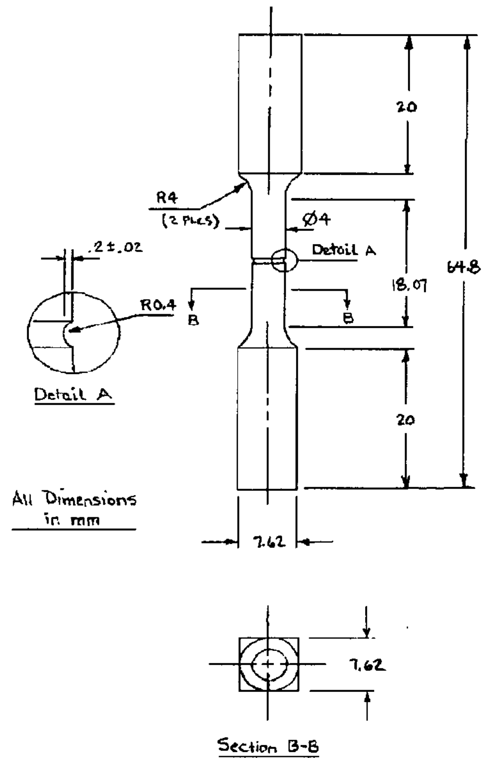

Figure G-12
SSRT Specimen for CIR Task 1 [16]

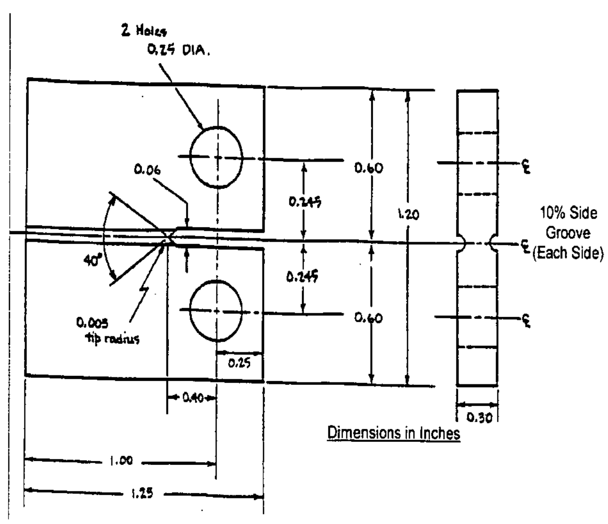

Figure G-13
Specimen Used for In-Situ Crack Growth Tests [16]

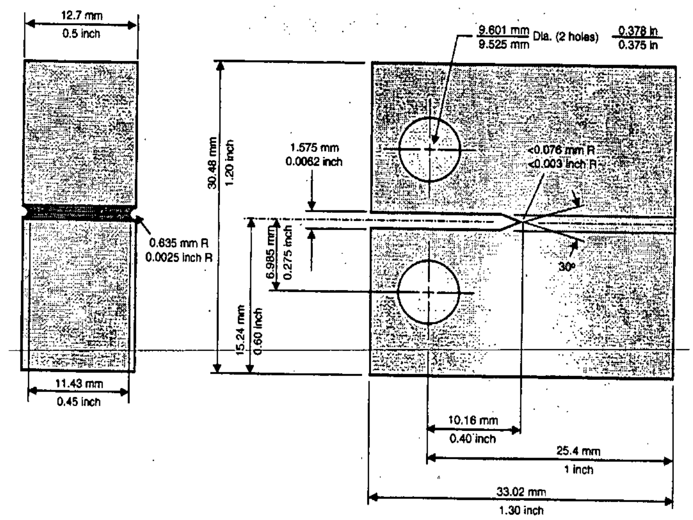

Figure G-14
Schematic Drawing of 0.5TCT Specimen [17]

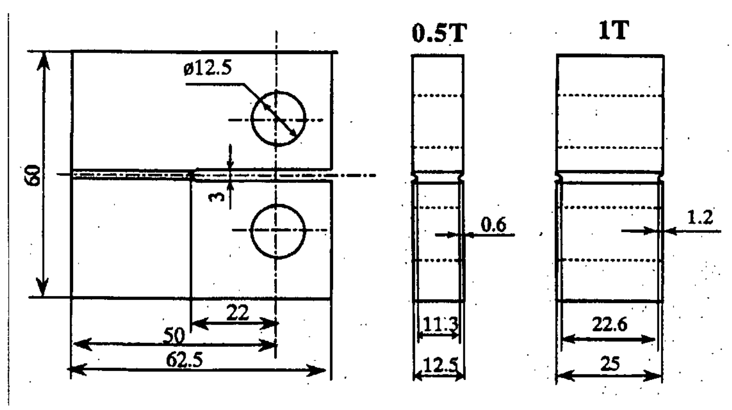

Figure G-15
Schematic Drawing of CT Specimens [18]

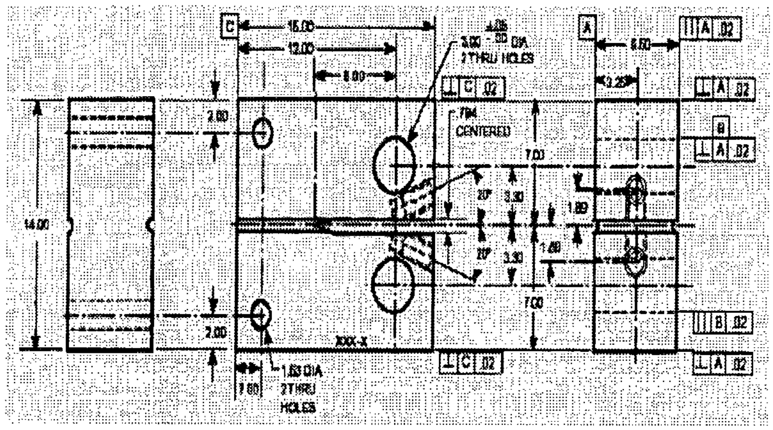

Figure G-16
CT Specimen Geometry [21]

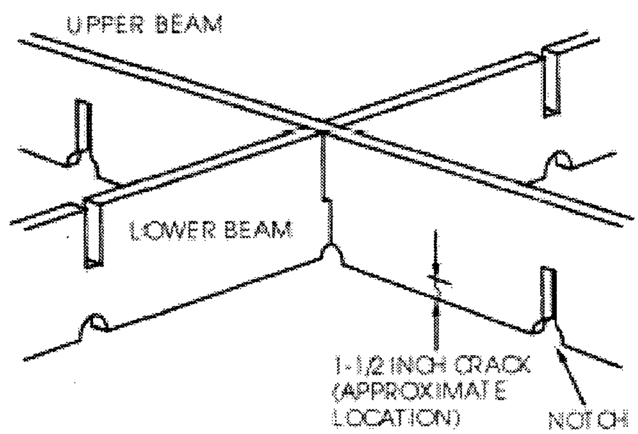

Figure G-17
Egg-Crate Design of Oyster Creek Top Guide (Showing Crack Location) [23]

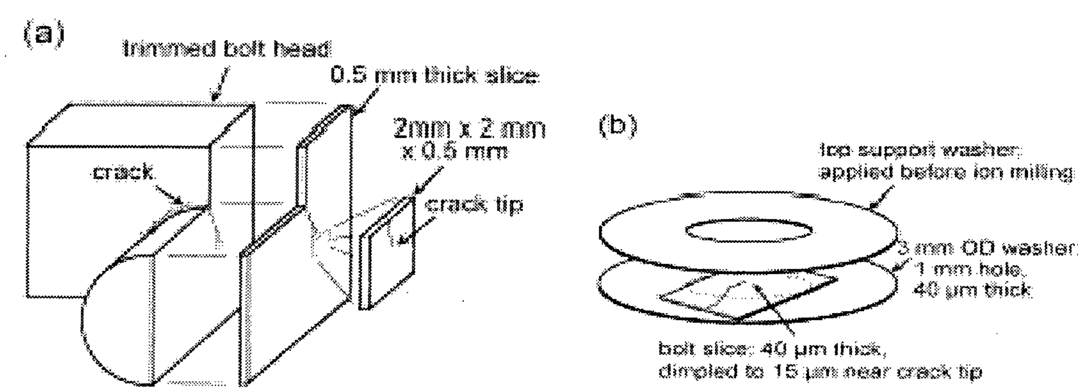

Figure G-18
TEM Sample Preparation from a Cracked Baffle Bolt [23]

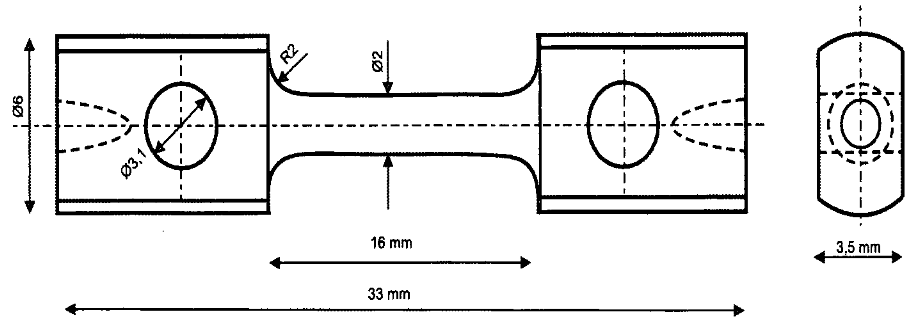

Figure G-19
Tensile Specimen Used for SSR Tests and Tensile Tests [25]

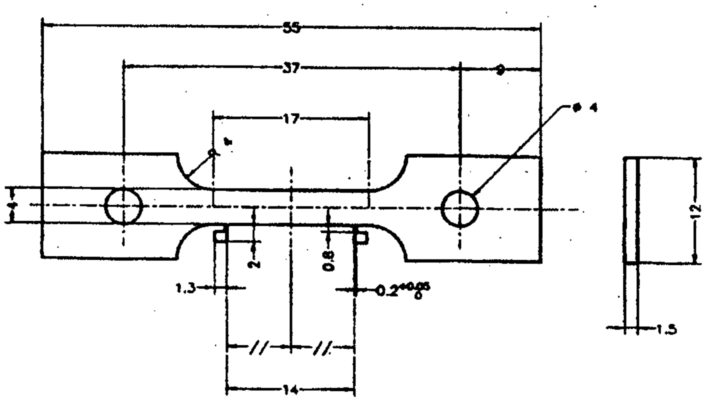

Figure G-20
Type of Tensile Specimen Used for Tensile Tests [27]

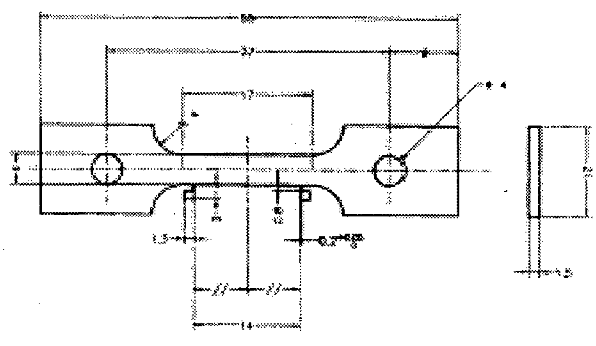

Figure G-21
Chooz A (JOBB) Tensile and Fracture Toughness Specimens [27, 28]

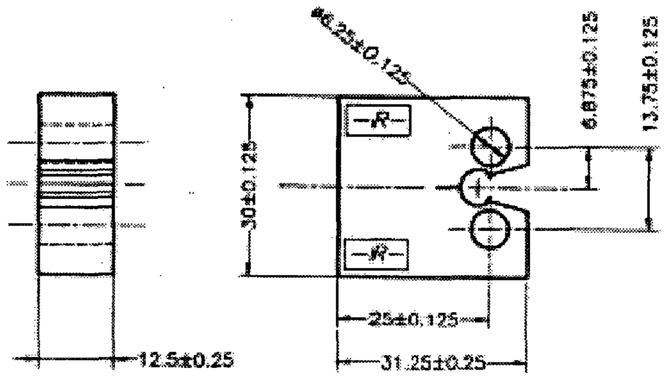

Figure G-22
Chooz A (JOBB) CT Specimen [28]

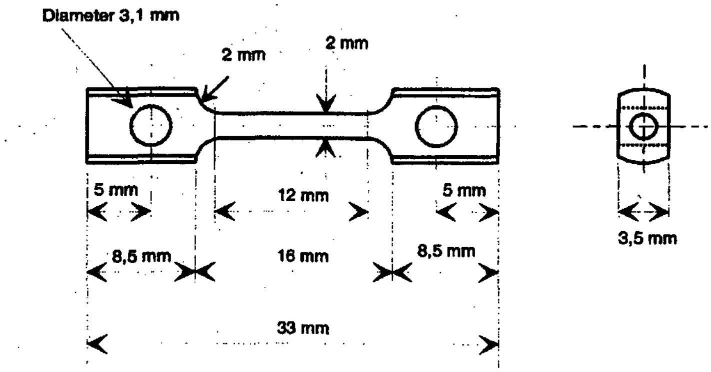

Figure G-23
Type of Tensile Specimen Used for Tensile Tests [30, 43, 49]

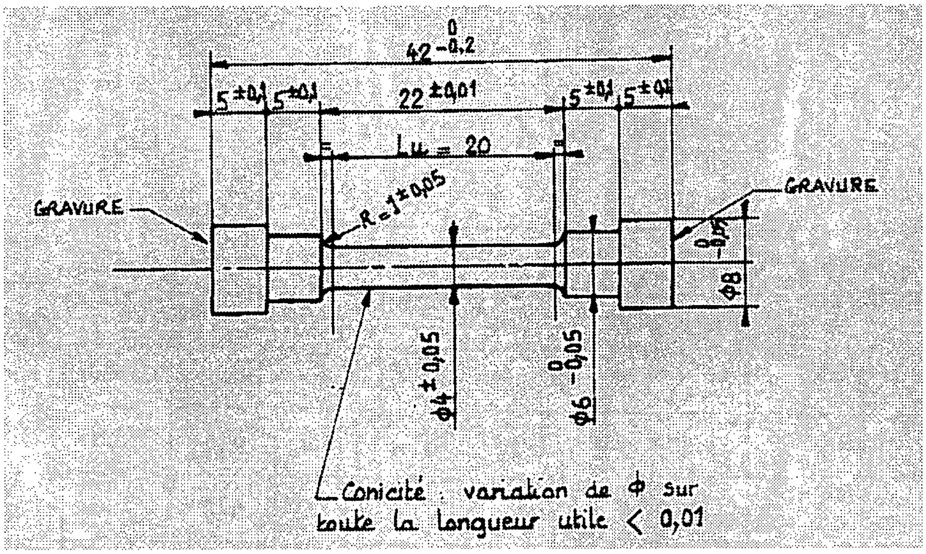

Figure G-24
Type of Tensile Specimen [38, 39]

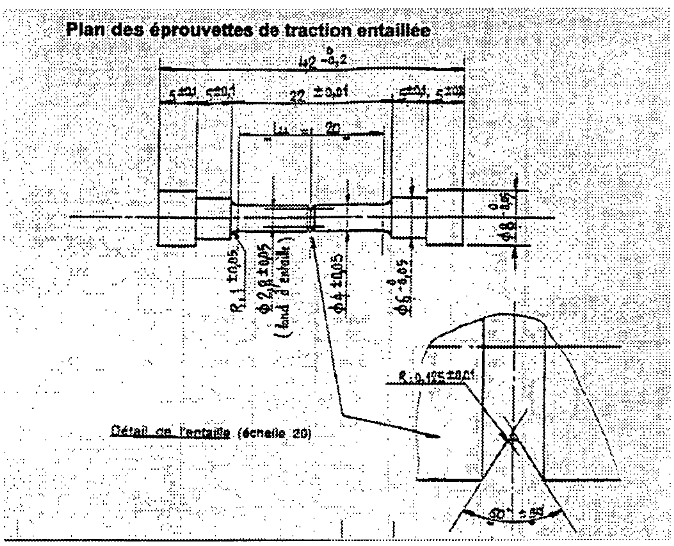

Figure G-25
Type of Tensile Specimen [39]

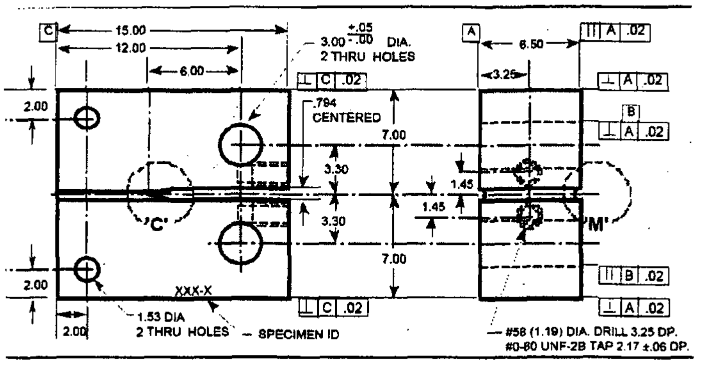

Figure G-26
1/4T CT Specimen Used By Chopra [45]

H
APPENDIX H: COMPOSITION

Table H-1
Chemical Composition of the Database Test Materials

Content deleted – EPRI/MRP Proprietary Information

I

APPENDIX I: REFERENCES (APPENDICES A-H)

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