NUREG-1511, Supp. 2

Reactor Pressure Vessel Status Report

Supplement 2

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
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ABSTRACT

On May 18, 1995, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," requesting that addressees: (1) identify, collect, and report any new data pertinent to the analysis of structural integrity for the reactor pressure vessels (RPVs) at their nuclear power plants, and (2) assess the impact of those data on their RPV integrity analyses relative to the requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and to the requirements of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," which is used to evaluate the upper shelf energy (USE) values for RPV materials and pressure-temperature (P-T) limits for RPVs.

Since the issuance of GL 92-01, Revision 1, Supplement 1, the industry completed a major initiative to collect all available alloying chemistry and materials property data for the various forging, plate and weld materials used to fabricate the RPVs. This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light-water nuclear power plants. These efforts should minimize surprises regarding the alloying chemistries for domestic RPV beltline materials. In addition, as a result of the industry's efforts in response to GL 92-01, Revision 1, Supplement 1, licensees and staff will be able to perform reactor vessel integrity evaluations more efficiently and more effectively. However, the staff expects additional surveillance data will become available after fracture toughness testing is performed on surveillance capsules that are presently being irradiated in domestic RPVs. The staff will incorporate the additional surveillance data into the existing database after it becomes available for review.

In the summer and fall of 1998, the staff issued a series of requests for additional information (RAIs) regarding the industry's responses to GL 92-01, Revision 1, Supplement 1. In the RAIs, the staff requested that the addressees assess how the updated alloying chemistry and materials property data would affect the results of the RPV integrity analyses for their plants.

This report summarizes both the industry's and the NRC's efforts to address how all of the new chemistry and surveillance data, when integrated, could affect the plant-specific RPV integrity analyses for the RPVs of U.S. light-water nuclear power plants. Specifically, this report discusses the following: (1) the basis for issuing and reviewing the responses to the RAIs on GL 92-01, Revision 1, Supplement 1, (2) the activities conducted by the industry owners groups to collect weld chemistry data for the materials used to fabricate the RPVs at U.S. nuclear plants, (3) the activities conducted by the industry owners groups to collect data from RPV surveillance capsules on behalf of the industry, (4) the staff's efforts to update plant-specific data into the Reactor Vessel Integrity Database, and to make Version 2 of the database accessible via the World-Wide-Web, (5) the staff's current regulatory and research activities regarding RPV integrity, (6) the staff's activities regarding RPV weld inspection reduction, as documented in the NRC final safety evaluation on Topical Report BWRVIP-05, (7) the staff's review of the thermal annealing project at the Marble Hill facility and the status of the Palisades thermal annealing application, and (8) the results of significant plant-specific RPV integrity reviews that could have an impact on the industry.

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EXECUTIVE SUMMARY

This NUREG describes the actions taken by the U.S. Nuclear Regulatory Commission (NRC), as well as nuclear industry owners groups and individual licensees, regarding the ongoing assessment of reactor pressure vessel (RPV) integrity. Following the issuance of Generic Letter (GL) 92–01, Revision 1, in March 1992 (Ref. 1), and NUREG–1511 in December 1994 (Ref. 2), the staff directed its efforts toward: (1) addressing an issue that some licensees were not aware of, or not using all of the available chemistry and surveillance data applicable to the evaluations of their RPV beltline materials, and (2) determining the generic implications of the larger-than-expected variability observed in the chemical compositions of RPV welds. To address these issues, the staff issued Supplement 1 to GL 92–01, Revision 1 (GL 92–01, Revision 1, Supplement 1), on May 18, 1995 (Ref. 3), and requested that the licensees to which the GL was addressed identify any new data that could be pertinent to, and impact the structural integrity analyses of, their RPVs relative to the requirements of Section 50.60 of Part 50 of Title 10, *Code of Federal Regulations* (Ref. 4), Section 50.61 of Part 50 of Title 10, *Code of Federal Regulations* (Ref. 5), and Appendices G and H to Part 50 of Title 10, *Code of Federal Regulations* (Refs. 6 and 7).[1]

Since the issuance of GL 92–01, Revision 1, Supplement 1, the industry owners groups (e.g., the Boiling Water Reactor Vessel and Internals Project (BWRVIP), the Combustion Engineering Owners Group (CEOG), and the Babcock and Wilcox Owners Group (B&WOG)) have completed a major initiative to collect all available alloying chemistry and materials property data for the various forging, plate and weld materials used in the fabrication of U.S. RPVs.

This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light-water nuclear power plants. These efforts should minimize surprises regarding the alloying chemistries for domestic RPV beltline materials. However, the staff expects that with the testing of future surveillance capsules, additional materials property and dosimetry data will become available. The Charpy materials property data and the reanalysis of capsule and RPV fluences (based on the dosimetry data) will be used to revise the existing database.

All licensees responded to GL 92–01, Revision 1, Supplement 1. Although some licensees provided additional data that were not included in their responses to the initial version of the GL, the collective responses to GL 92–01, Revision 1, Supplement 1, demonstrated that there were no new RPV integrity issues that would be considered as an immediate safety concern. The majority of the licensees also indicated that they were participating in the activities of their respective industry owners groups to collect and analyze available RPV weld chemistry (specifically the copper and nickel contents) and surveillance data on behalf of their member utilities. The owners groups have now completed these initiatives.

In the summer and fall of 1996, NRC staff issued closeout letters on GL 92–01, Revision 1, Supplement 1, which noted that no immediate safety issues were associated

[1] Henceforth, Sections of Part 50 to Title 10 of the *Code of Federal Regulations* will be abbreviated 10 CFR 50.XX or 10 CFR 50.XXX. Appendices to Part 50 will be abbreviated as 10 CFR Part 50, Appendix X designations.

with the structural integrity assessments for U.S. light-water reactors (LWRs). However, the staff also acknowledged that ongoing RPV initiatives were being conducted by the industry owners groups, and informed the licensees that additional NRC work might be scheduled pending its review of the results of these initiatives.

This review of the industry's RPV integrity initiatives led the NRC staff to conclude that the new data compiled by the owners groups could have an impact on the pressurized thermal shock (PTS) and pressure-temperature (P-T) limit assessments for some facilities; however, the staff did not consider this to be an immediate safety concern because these assessments are considered to be time dependent analyses.[(2)] As a result, the staff concluded that there was sufficient time for licensees to assess the impact of the new data on their PTS assessments (applicable to PWRs only) and P-T limit assessments prior to the expiration date of the operating licenses for their facilities.[(2)] Therefore, in 1998, the staff issued a series of requests for additional information (RAIs) to the majority of licensees that responded to GL 92–01, Revision 1, Supplement 1, and requested that the recipients assess the impact of the newly compiled chemistry and surveillance data on the PTS and P-T limit evaluations for their facilities. This report, in part, summarizes the staff's evaluation of the responses to the RAIs on GL 92–01, Revision 1, Supplement 1. The status of the staff's evaluations through its review of the responses to the RAIs on GL 92–01, Revision 1, Supplement 1, indicates that, with the exception of the RPVs for the Palisades and Fort Calhoun nuclear plants, all RPVs of light-water reactors in the U.S. will be in compliance with the requirements of 10 CFR Part 50.61, and 10 CFR Part 50, Appendices G and H, throughout the terms of the operating licenses for the facilities. Both the licensees for the Palisades and Fort Calhoun nuclear plants have ongoing efforts to review the materials property and neutron fluence data for their RPVs, and to address compliance with the PTS requirements of 10 CFR 50.61.

Several developments have also occurred in the area of Codes and Standards activities and research developments that affect RPV integrity assessments. In particular, the American Society of Mechanical Engineers (ASME) Code Committees have passed Code Cases N–640 and N–588 (Refs. 9 and 10), which have direct consequences on P-T limit assessments, and Code Case N–629 (Ref. 11), which proposes a new methodology (the "Master Curve") for indexing material reference temperatures. Research developments have included work on new RPV embrittlement correlations, RPV flaw distribution studies, ongoing work to support the Master Curve methodology, and an effort by the NRC and industry to revise the PTS screening criteria.

The staff has also considered proposals to reduce the scope of augmented inspections performed on the circumferential shell welds of boiling water reactor (BWR) RPVs. The basis for this proposed reduction in scope for the augmented inspections, which are required pursuant to paragraph (g)(6)(ii)(A) of 10 CFR 50.55a (Ref. 12), was proposed by the BWRVIP and is documented in Topical Report BWRVIP–05 (Ref. 13). The staff approved this topical report on July 28, 1998 (Ref. 14), and stated its position for reduced

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(2) The staff's generic assessment is documented in a memorandum from Jack R. Strosnider, Chief, Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, to Ashok C. Thadani, Associate Director for Technology, Office of Nuclear Reactor Regulation, dated May 5, 1995. This memorandum was included as part of Commission Paper SECY-95-119, "Status of Reactor Pressure Vessel Issues" (May 8, 1995, Ref. 8)

inspection scopes of BWR circumferential RPV shell welds in GL 98–05 (Ref. 15), which was issued on November 10, 1998. The staff has currently approved a number of proposals for reduced inspections of the circumferential RPV shell welds in BWR-designed facilities.

With the adoption of 10 CFR 50.66 (Ref. 16), the staff has also established a regulatory framework for the thermal annealing of RPVs. Guidance for complying with 10 CFR 50.66 is documented in RG 1.162 (Ref. 17). The feasibility of thermal annealing was demonstrated by a joint Department of Energy (DOE)/Industry-sponsored annealing demonstration project (ADP); this ADP was performed at a decommissioned Marble Hill reactor unit, and applied indirect heating as the method for annealing the unit's RPV. A second ADP using an electrical resistance heating approach was terminated.

The staff has also updated the reactor vessel integrity database (RVID). The RVID provides an efficient and effective means of storing and maintaining RPV vessel data relative to the requirements of 10 CFR 50.60; 10 CFR 50.61; and 10 CFR Part 50, Appendices G and H. The database can be used as an effective means of indicating compliance with these regulations and maintains safety through a comprehensive and integrated approach. RVID Version 2 (RVID 2) was issued on the World-Wide-Web in June 1999. RVID 2 is a Windows 3.1 native application based on Microsoft Access 2.0™.

The four RVID 2 diskettes can be downloaded from the homepage at (http://www.nrc.gov/NRR/RVID/index.html) which is linked to the NRC homepage. RVID was developed following the staff's review of licensee responses to GL 92–01, Revision 1. The database summarizes the properties of the reactor vessel beltline materials for each operating commercial nuclear power plant. RVID 2 reflects licensee responses to the RAIs on GL 92–01, Revision 1, Supplement 1. The database will be updated when sufficient amounts of new surveillance data, chemistry data, or fluence evaluations warrant a new revision.

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ABBREVIATIONS

10 CFR	Title 10 to the *Code of Federal Regulations*
ADP	annealing demonstration project
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock and Wilcox Company, now Framatome Technologies, Inc.
B&WOG	Babcock and Wilcox Owners Group
BWR	boiling water reactor
BWR-4	boiling water reactor 4 design, one of General Electric's model designs for boiling water reactors
BWROG	Boiling Water Reactor Owners Group
BWRVIP	BWR Vessel and Internals Project
CB&I	Chicago Bridge and Iron Works
CDF	core damage frequency
CE	Combustion Engineering Corporation, which is currently named CE Nuclear Power LLC
CEOG	Combustion Engineering Owners Group
CF	chemistry factor
CPCo	Consumers Power Company
CRD	control rod drive
DOE	U.S. Department of Energy
EOL	end of license
EPRI	Electric Power Research Institute
GL	Generic Letter
HPCI	high pressure coolant injection
HPCS	high pressure coolant spray
IHI	Ishikasajima-Hirama Heavy Industries
IN	Information Notice
ISA	independent safety assessment
ISI	inservice inspection
LPCI	low pressure coolant injection
LPCS	low pressure coolant spray
LTOP	low temperature overpressure protection
LWR	light water reactor
MOU	memorandum of understanding
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORNL	Oak Ridge National Laboratories
P(FIE)	conditional probability of failure
P-T	pressure-temperature
PFM	probabilistic fracture mechanics

PNNL	Pacific Northwest National Laboratories
PTS	pressurized thermal shock
PVRUF	Pressure Vessel Research Users Facility
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCIC	reactor core isolation cooling
RES	Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission
RFO	refueling outage
RG	Regulatory Guide
RPV	reactor pressure vessel
RVID	Reactor Vessel Integrity Database
RVID 2	RVID Version 2
SER	safety evaluation report
SLC	standby liquid control
TAR	thermal annealing report
USE	upper shelf energy
UT	ultrasonic testing
VTT	Technical Research Center of Finland (Valtion Teknillinen Tutkinuskeskus)

MATHEMATICAL AND SCIENTIFIC NOMENCLATURE

A_i — as defined in equation 5 of 10 CFR 50.61, the measured ΔRT_{NDT} value for base metal materials, or for weld materials, the value of ΔRT_{NDT}, as adjusted to account for the differences in the table CFs for the RPV and surveillance capsule weld materials

ART — adjusted reference temperature

°C — abbreviation for degrees C, a standard unit of temperature in the Centigrade temperature scale

CF — chemistry factor, which is a function of the copper and nickel alloying contents of reactor pressure vessel materials and is used in the calculations of ART and RT_{PTS}

Cu — Periodic Table abbreviation for the element copper

°F — abbreviation for degrees F, a standard unit of temperature in the Fahrenheit temperature scale

f — projected neutron fluence value for a RPV material for neutrons having kinetic energies in excess of 1 MeV - reported in units of 10^{19} n/cm^2 (10^{19} neutrons per square centimeter)

f_i — as defined in equation 5 of 10 CFR 50.61, the neutron fluence value for the A_i surveillance data point

ft-lb — an abbreviation for foot-pounds, a standard unit of work or energy in the English system of weights and measures

joules — a standard unit of work or energy in the SI system of weights and measures

K_{1a} — lower bound crack arrest fracture toughness from Section XI of the ASME Code

K_{1c} — lower bound static initiation fracture toughness from Section XI of the ASME Code

M — margin term to be added in the calculations of adjusted reference temperatures to account for uncertainties in the calculational procedures, the initial reference temperature, the copper and nickel contents of the vessel material, and the neutron fluence

Mn — Periodic Table abbreviation for the element manganese

Mo — Periodic Table abbreviation for the element molybdenum

MPa — an abbreviation for megapascals, a unit of pressure or stress in the SI system of weights and measures

n — as defined in equation 5 of 10 CFR 50.61, the number of surveillance capsule data points

Ni — Periodic Table abbreviation for the element nickel

P(FIE) — conditional probability of failure, a parameter used in probabilistic fracture mechanics calculations for reactor pressure vessel materials psig — an abbreviation for pounds per square inch, a unit of pressure or stress in the English system of weights and measures

RT_{NDT} — the reference temperature for a RPV material

$RT_{NDT(U)}$ — the initial reference temperature of the RPV material in the unirradiated condition

RT_{PTS} — the reference temperature for pressurized thermal shock, which is equivalent to the adjusted reference temperature for a RPV material in PWRs at the end-of-license for the facility, as determined using the best estimate end-of-license neutron fluence for the material at the clad-base metal interface of the vessel

RT_{T0} — as defined in ASME Code Case N-629, the fracture toughness curve indexing temperature based on use of the Master Curve methodology which replaces the use of RT_{NDT} (mathematically defined as equal to T_0 + 35°F)

Table $CF_{vessel\ chem.}$ — CF for a particular reactor vessel weld as determined from the tables in 10 CFR 50.61, and based on its copper and nickel alloying contents

Table $CF_{vessel\ surv.}$ — CF for a particular surveillance capsule weld material as determined from the tables in 10 CFR 50.61, and based on its copper and nickel alloying contents

T_0 — As defined in ASTM Standard Procedure E1921, and used in Master Curve methodology, a temperature defined to correspond to a fracture toughness of 100 MPa√in

ΔRT_{PTS} — the specific term for the mean value of the increase (shift) in the reference temperature for materials in PWR RPVs as a result of the effects of neutron irradiation on the materials, as used in the calculations for pressurized thermal shock (i.e., RT_{PTS} calculations)

ΔRT_{NDT} — the general term for the mean value of the increase (shift) in the reference temperature for RPV materials as a result of the effects of neutron irradiation on the materials

$\sum$ — Symbol for a summation function in mathematics

σ_U — standard deviation term for $RT_{NDT(U)}$, a parameter used in the calculation of margin terms (M) in plant specific evaluations for pressurized thermal shock (for the case of general ART calculations, this term is referred to as σ_I)

σ_Δ — standard deviation term for ΔRT_{NDT} or ΔRT_{PTS}, a parameter used in the calculation of margin terms (M)

1 INTRODUCTION

1.1 Overview

The Nuclear Regulatory Commission (NRC) has established regulations to address the implications of accumulated neutron irradiation on the structural integrity of the RPVs in the commercial nuclear industry. These regulations include 10 CFR 50.60 (Ref. 4); 10 CFR 50.61, the Pressurized Thermal Shock (PTS) Rule (Ref. 5); and 10 CFR Part 50, Appendices G and H (Refs. 6 and 7). 10 CFR 50.60 requires licensees to comply with the reactor coolant pressure boundary requirements and RPV material surveillance program requirements set forth in 10 CFR Part 50, Appendices G and H, respectively. 10 CFR 50.60, however, allows licensees to use an alternative to the requirements of 10 CFR Part 50, Appendices G and H, when the Commission grants an exemption under the requirements of 10 CFR 50.12. Both 10 CFR 50.61, and 10 CFR Part 50, Appendix G, establish limits on the degree to which the RPV may be embrittled as a result of neutron irradiation. Another regulation, 10 CFR Part 50, Appendix H, establishes the requirements for developing plant-specific RPV surveillance data that are used to monitor the structural integrity assessments required by 10 CFR 50.61, and 10 CFR Part 50, Appendix G.

1.2 The Pressurized Thermal Shock Rule

10 CFR 50.61, the PTS Rule, defines screening criteria for embrittlement of RPV materials in pressurized-water reactors (PWRs), as well as the actions that are required if these screening criteria are exceeded. The screening criteria limit the degree that a vessel material may increase in its reference temperature (RT_{PTS}) following neutron irradiation of the RPV. The RT_{PTS} values, which are based on the projected end-of-license (EOL) neutron fluence values for the RPV materials, are calculated in accordance with Equation 1–1:

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + M \qquad (1\text{-}1)$$

In this equation, $RT_{NDT(U)}$ represents the initial reference temperature of the vessel material in the unirradiated condition; ΔRT_{PTS} represents the increase (shift) in reference temperature value for the material as a result of neutron irradiation of the RPV; and M represents the margin to be added in the calculations to account for uncertainties in the calculational procedures and in the methods for measuring the initial reference temperatures, copper and nickel contents, and neutron fluence values. Specifically, $M = 2\sqrt{(\sigma_U^2 + \sigma_\Delta^2)}$, where σ_U is the standard deviation for $RT_{NDT(U)}$ and σ_Δ is the standard deviation for ΔRT_{PTS}. The PTS rule requires licensees to calculate the shift in an RPV material's reference temperature value (ΔRT_{PTS}) in accordance with Equation 1-2:

$$\Delta RT_{PTS} = CF \times f^{(0.28 - 0.10 \times \log(f))} \qquad (1\text{-}2)$$

In Equation 1-2, "f" represents the projected EOL neutron fluence (in units of 10^{19} n/cm^2, for neutrons with kinetic energies greater than 1 MeV) for the material at the clad-base metal interface (i.e., at the inside surface of the base metal) for the vessel, and CF presents a proportionality factor, otherwise known as the chemistry factor. The PTS rule requires the CF for an RPV material to be determined by one of two methods:

(1) by applying the methods of section (c)(1)(iv)(A) to 10 CFR 50.61, which gives the bases for using the tables in the rule to calculate the CF as a function of the copper and nickel alloying (chemistry) contents of the material; or

(2) by applying the methods of sections (c)(2)(ii and iii) to 10 CFR 50.61 and the results of Charpy-V impact tests on surveillance capsule specimens removed in accordance with a utility's reactor vessel material surveillance program (i.e., the .10 CFR Part 50, Appendix H, program for the plant) if the testing data have been determined to be credible in accordance with the criteria of 10 CFR 50.61(c)(2)(i)(A–E).

The screening criteria in the PTS rule are 132°C (270°F) for plate, forging, and axial weld materials and 149°C (300°F) for circumferential weld materials. When RT_{PTS} values are projected to exceed these screening criteria, the rule requires that licensees perform neutron flux reductions, plant modifications, or additional plant-specific evaluations of their RPVs to justify continued operation of their reactors.

1.3 Requirements for Upper Shelf Energy and Pressure-Temperature Limits

10 CFR Part 50, Appendix G, contains screening criteria that limit the degree that an RPV material may drop in its upper shelf energy (USE) value following irradiation of the vessel. The regulation requires the USE for an RPV material (as measured from the results of Charpy-V impact tests) to be greater than 102 joules (75 ft-lb) when the material is in the unirradiated condition. The regulation also requires the USE of the material to remain above 68 joules (50 ft-lb) throughout the licensed life of the vessel. If these conditions are not met, the regulation requires that additional fracture mechanics analyses be performed to demonstrate that sufficient margins of safety will exist for lower values of USE. These safety margins must be at least as conservative as those that would be obtained if the criteria in the edition and addenda of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Appendix G to the ASME Code, Ref. 18), as endorsed in 10 CFR 50.55a (Ref. 12), were used to satisfy the safety margin requirements.

Through efforts coordinated by the owners groups, the industry has performed equivalent margins analyses to demonstrate that USE values below 68 J (50 ft-lb) will provide margins of safety against fracture equivalent to those required by Appendix G to the ASME Code. These analyses were performed for generic groupings of plants. In addition, some licensees performed plant-specific equivalent margins analyses. In NUREG/CR-6023 (Ref. 19), the NRC staff concluded that PWR and boiling water reactor (BWR) RPV materials could have EOL USE values less than 68 joules (50 ft-lb) and still provide the required margins of safety against fracture. On the basis of the industry's equivalent margins analyses and NRC's generic study, the staff concluded in NUREG-1511 (Ref. 2) that all RPVs will have adequate upper-shelf toughness throughout their current licensed operating life.

10 CFR Part 50, Appendix G, also establishes requirements for the calculation of P-T limit curves that are used as a means of protecting the integrity of the RPV during normal operating conditions, anticipated operational occurrences, and pressure testing conditions. These P-T limits are used to establish low-temperature overpressure protection (LTOP) system setpoints for the plants. The regulation requires that the P-T limits must be at least as conservative as those that would be generated using the

methods of analysis and margins of safety of Appendix G to the ASME Code, as endorsed in 10 CFR 50.55a. 10 CFR 50.60 requires that licensees submit requests for exemptions to use less conservative P-T limits when this condition is not satisfied. Such requests are evaluated, and granted or denied by NRC, on a case-by-case basis.

1.4 Generic Letter 92–01, Revision 1, NUREG–1511, and Supplements

The "Reactor Pressure Vessel Status Report," NUREG–1511 (Ref. 2), described the RPV and discussed the effect of radiation embrittlement on RPV materials and the indicators for measuring embrittlement. NUREG–1511 also summarized the results of the NRC staff's review of the industry's responses to Generic Letter (GL) 92–01, Revision 1 (Ref. 1), as well as plant-specific RPV evaluations for the 37 BWR plants and 74 PWR plants in the United States. The data resulting from the staff's review are stored in NRC's RVID database. Following the issuance of GL 92–01, Revision 1, and NUREG–1511, the staff directed its efforts toward assuring that the licensees for U.S. light-water reactors were evaluating their RPVs by applying all available chemistry and surveillance data pertinent to the RPV assessments required by the confines of the current regulatory framework (e.g., within the regulatory framework of 10 CFR 50.60; 10 CFR 50.61; and 10 CFR Part 50, Appendices G and H). To address this concern, the staff issued Supplement 1 to GL 92–01, Revision 1 (Ref. 3), on May 18, 1995. In this supplement to the GL, the staff requested that the addressees identify, collect, and report any new data that could be pertinent to, and impact the structural integrity analyses of their RPVs relative to the requirements of 10 CFR 50.60; 10 CFR 50.61; and 10 CFR Part 50, Appendices G and H.

All licensees responded to GL 92–01, Revision 1, Supplement 1. Although some licensees provided additional data that were not included in their responses to the initial version of the generic letter, all licensees indicated that there were no new RPV integrity issues that would have immediate safety concerns. The majority of the licensees also indicated that they were participating in the activities of their respective industry owners group (e.g., BWRVIP, CEOG, or B&WOG) to collect and analyze available RPV weld chemistry data (specifically, copper and nickel content data) and surveillance data on behalf of its member utilities. The owners groups have now completed these initiatives. This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light-water nuclear power plants. These efforts should minimize surprises regarding the alloying chemistries for domestic RPV beltline materials. However, the staff expects additional surveillance data will become available after fracture toughness testing is performed on surveillance capsules that are presently being irradiated in domestic RPVs. The staff will incorporate the additional surveillance data into the existing database after it becomes available for review.

In October 1996 the NRC staff issued Supplement 1 to NUREG–1511 (Ref. 20). The updated "Reactor Pressure Vessel Status Report" discussed: (1) the basis for issuing GL 92–01, Revision 1, Supplement 1; (2) the status of licensee Responses to GL 92–01, Revision 1, Supplement 1; (3) the up-to-date status of licensee compliance with the PTS rule; (4) the NRC's establishment of a framework for the thermal annealing of RPVs; and (5) the staff's development of an updated version of the RVID. In the summer and fall of 1996, the staff issued its closeout letters on

GL 92–01, Revision 1, Supplement 1. In these close-out letters, the staff noted that no immediate safety issues were associated with the structural integrity assessments for U.S. LWRs. However, the NRC staff also acknowledged that ongoing RPV initiatives were being conducted by the industry owners groups and informed the licensees that additional NRC work might be scheduled pending its review of the results of these initiatives.

This review of the industry's vessel integrity initiatives led the NRC staff to conclude that the new data compiled by the owners groups could have an impact on the PTS and P-T limit assessments for some facilities. Therefore, in 1998, the staff issued a series of RAIs to the majority of licensees that responded to GL 92–01, Revision 1, Supplement 1, and requested that the recipients assess the impact of the new chemistry and surveillance data on the PTS and P-T limit evaluations for the facilities. Chapter 2 of this report, in part, summarizes the staff's evaluation of the responses to the RAIs on GL 92–01, Revision 1, Supplement 1.

2 GENERIC LETTER 92–01 REVISION 1, SUPPLEMENT 1 : REQUESTS FOR ADDITIONAL INFORMATION

2.1 Background

The publication of Supplement 1 to NUREG–1511 (Ref. 20) in October 1996 and the issuance of administrative closeout letters to licensees in late 1996 and early 1997 signified the administrative completion of the NRC staff's review of the industry's responses to GL 92–01, Revision 1, Supplement 1 (Ref. 3). At that time, some of the owners groups had ongoing activities to collect the best estimate chemistry values (specifically copper and nickel content data) and surveillance data for RPV weld materials. This data is important for the estimation of the fracture toughness properties of the materials, and particularly to the application of the chemistry factor ratio procedure methodology described in Position 2.1 of RG 1.99, Revision 2 (Ref. 21). Since that time, the owners groups have completed these activities and have submitted a number of topical reports to the staff which summarize these activities (Refs. 22-26). In addition, the NRC staff has conducted an inspection of Framatome Technologies, Inc. (FTI), to obtain all available RPV weld chemistry data for welds fabricated by Babcock and Wilcox (B&W). To ensure that licensees have considered the impact of these activities on their RPV integrity analyses, the NRC staff issued requests for additional information (RAIs) to specific licensees during 1998.

This chapter describes industry and NRC activities relative to the nuclear industry's collection of best estimate RPV weld chemistry and RPV material surveillance data, and the issuance of the staff's RAIs. The discussion of the RAIs includes a description of the content of the RAIs and the status of the NRC's review of the licensee responses to them.

2.2 Activities Regarding RPV Weld Chemistries

As a part of GL 92–01, Revision 1, Supplement 1, the NRC staff requested that licensees provide "a description of those actions taken or planned to locate all data relevant to the determination of RPV integrity . . . " The NRC staff's request was a direct result of observations that some licensees were not aware of or not using all the available chemistry and surveillance data applicable to the evaluations of their RPV beltline materials, as required by 10 CFR 50.61 and 10 CFR Part 50, Appendices G and H. The staff made this observation after noticing that different licensees had reported significantly different copper and nickel content values (also referred to in this report as "chemistry values") for RPV welds made from the same heat of weld wire. The responses from the individual licensees to GL 92–01, Revision 1, Supplement 1, indicated that several owners group activities would be initiated to collect, analyze, and, in some cases, evaluate the impact of assessing all relevant RPV weld chemistry and surveillance data. The staff's RAIs were issued in part as a means of following up on a particular licensee's commitment to assess the impact of these owners group activities on the plant-specific RPV evaluations for its facility.

Actions to address GL 92–01 issues were undertaken by three of the owners groups: B&WOG, CEOG, and BWRVIP. The latter is a technical group under the auspices of the Boiling Water Reactor Owners Group

(BWROG). These owners group activities culminated in the submission of several final reports to the staff. Individual licensees then referenced these reports in their docketed responses to the RAIs.

The first report received from the CEOG was Topical Report CE NPSD–1039, Revision 2 (Ref. 22). In this report, the CEOG described the methodologies used to assimilate all of the data on Combustion Engineering (CE) weld materials, to evaluate and screen data points, and to calculate the best estimate copper and nickel contents for each weld wire heat (or tandem wire combination) addressed in the report. The NRC staff examined the CEOG report, generally agreed with the approach taken, and developed comments on the methodologies proposed by the CEOG. These comments served as the basis for the staff's presentations in meetings with industry owners groups on GL 92–01-related topics (Refs. 27 and 28). The staff's comments were also formally forwarded to the CEOG (Ref. 29).

The NRC's comments on CE NPSD–1039, Revision 2, are briefly discussed below and can be grouped into three broad categories: (1) the methods proposed for screening outliers from the weld wire chemistry database, (2) the use and development of generic best estimate chemistry values for a class of weld materials, and (3) the choice of a best estimate computational methodology based on the information available for a specific weld wire heat. The NRC staff emphasized that when proposing to exclude outlier data points from best estimate chemistry value evaluations, both statistical and physical bases should be provided to demonstrate that the data points should not be included in the evaluation data. On the use of generic best estimate chemistry values, the NRC staff affirmed its position that a single valid data point could serve as the basis for the determination of weld wire heat specific best estimate chemistry values, but that when such limited data were cited, the NRC staff would consider the impact of using the generic chemistry values for the appropriate material class. This was done to provide confidence that sufficient margins existed in the RPV assessments in light of the variability reported in RPV weld chemistries. Finally, the NRC staff also noted that, in general, the use of coil-weighted or group-weighted averages for determining the best estimate chemistry values for a weld wire heat was preferable to the use of simple averages when chemistry data from several sources of data for that heat existed. A complete description of the meaning of *simple*, *coil-weighted*, and *group-weighted* averages is provided in CE NPSD-1039, Revision 2. It is sufficient to say here that if chemistry data points existed from a number of surveillance welds for the same weld wire heat:

- The *simple average* best estimate composition is calculated by adding up each individual data point and dividing the sum by the total number data points.

- The *group-weighted average* best estimate composition is calculated by using the average chemistry values for each of the surveillance welds and averaging the sum of the individual average values.

- The *coil-weighted average* best estimate composition is calculated by using the average chemistry values for each of the surveillance welds, multiplying them by the number of weld wire coils used in each of the welds, totaling the resultant values, and dividing the sum by the total number of coils.

As a result of NRC staff comments, the CEOG reevaluated some of the data reported in CE NPSD–1039, Revision 2.

In July 1998, the CEOG provided a revised report to the NRC staff, Topical Report CE NPSD–1119, Revision 1 (Ref. 23).

In this report, the CEOG supplied specific answers to comments presented in the NRC staff's letter of March 27, 1998 (Ref. 29), addressed questions that had been raised by the staff in the plant-specific RAIs, and provided a revised summary of best estimate values for CE RPV weld wire heats (and tandem wire combinations). As such, Topical Report CE NPSD–1119, Revision 1, provides the final and most complete evaluation of weld chemistry values for CE fabricated vessels.

Regarding the work by the B&WOG, initial evaluations of Babcock and Wilcox (B&W) RPV weld chemistry values were received from FTI in June and July of 1997 after the NRC had inspected the RPV data available at FTI's Lynchburg, Virginia, facilities (Ref. 30). These initial evaluations provided the raw B&W weld chemistry data and simple average best estimate chemistry values for B&W fabricated RPV welds. Subsequently, in May 1998, the B&WOG submitted a more in-depth report, BAW–2325, Revision 0. The B&WOG supplemented this report with the submittal of Topical Reports BAW–2325, Supplement 1 (Ref. 24), and BAW-2325, Revision 1 (Ref. 25), which were issued to incorporate comments and recommendations from the NRC staff.

The approach taken by the B&WOG in the BAW–2325 reports differed somewhat from that taken by the CEOG in the CE NPSD–1039 and CE NPSD–1119 reports. At the November 1997 and February 1998 meetings between the NRC staff and the industry, the staff not only addressed the evaluation of best estimate chemistry values, as mentioned previously, but also the evaluation of Charpy-based RPV surveillance data. Therefore, the B&WOG reports differed from the CE report in that it also included the evaluation of surveillance data for B&WOG facilities and provided, for each licensee, an assessment of the data's impact on the RPV integrity assessments for its plant(s). The evaluations of the B&W best estimate weld chemistry values were based on the use of the group-weighted average methodology. The NRC staff did not raise any additional questions regarding the methodology used for evaluating the best estimate weld chemistry values cited in the B&WOG report.

The final owners group assessment of RPV weld chemistry values was performed by the BWRVIP and submitted in Topical Report BWRVIP–46 (Ref. 26). In this report, the BWRVIP assessed the impact of new copper and nickel chemistry data on the RPV integrity assessments for the boiling water reactor (BWR) facilities. Since a large subset of the BWR vessels were manufactured by CE and B&W, much of the raw chemistry data had already been compiled and assessed in the CE and B&W reports. Additional data from the other principal BWR RPV manufacturer, Chicago Bridge and Iron (CB&I), were also reported and evaluated in Topical Report BWRVIP–46.

The approach to assess the impact of the newly compiled chemistry data on BWR vessel integrity in the BWRVIP–46 report was also different from the approaches taken by the B&WOG and CEOG. The basis for the report was to examine the best estimate values of copper and nickel and the range of copper and nickel contents for the RPV weld material having the highest (most limiting) adjusted reference temperature (ART), as reported for each BWR RPV. If the copper variability for a limiting material was less than a characteristic value (i.e., 0.05 percent, associated with the copper variability assumed in the PTS rule basis), no additional evaluation was performed. It was determined that the reported values for materials with less than 0.05 percent copper variability would be sufficient to ensure that pressure-temperature (P-T) curves were adequate for RPV operation. If the variability was greater than 0.05 percent, then the upper bound data point from all of the chemistry data available for the limiting RPV weld wire heat was evaluated to determine whether the ART calculated from the use of that data point would indicate that the current basis for the licensee's P-T limit curves was non-conservative. This approach indicated that one insignificant change would occur for a single BWR licensee, a 2°C (3°F) change in the ART for Cooper Nuclear Station, from 39°C to 41°C (102°F to 105°F). The NRC staff reviewed the BWRVIP–46 report, confirmed the acceptability of the approach taken therein, and issued a letter (Ref. 31) to the BWRVIP accepting the report and closing out the staff's review.

Since the information submitted in these reports was supplied by the owners groups and thus not directly associated with the licensing docket of any specific licensee, the NRC staff requested in its RAIs on GL 92-01, Revision 1, Supplement 1, that licensees review the information in these reports and determine its applicability to their facilities. Information submitted by the licensees in their responses to the RAIs consistently referenced the owners groups topical reports or the data presented in them. Therefore, the staff concluded that, as a result of the staff's GL 92–01 initiative and the owners groups activities in response to GL 92–01, a consistent industry-wide basis for the establishment of RPV weld best estimate chemistry values had, for the first time, been established.

This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light-water nuclear power plants. These efforts should minimize surprises regarding the alloying chemistries for domestic RPV beltline materials.

2.3 Activities Regarding RPV Surveillance Data

As a part of GL 92–01, Revision 1, Supplement 1, the NRC staff requested that licensees provide "a determination of the need for use of the ratio procedure in accordance with Position 2.1 of RG 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation." The ratio procedure cited in the request refers to adjustments to the measured ΔRT_{NDT} surveillance data to account for variances in the chemical compositions (specifically the copper and nickel contents) of the surveillance weld and the vessel weld. The staff included this request in the GL in order to assure that, if the best-estimate chemistry values for the RPV materials changed as a result of the newly reported chemistry and surveillance data, and were significantly different from the chemistry values for the surveillance specimens, the effects of the new data would be appropriately accounted for in the plant's PTS assessments and P-T limit curve calculations. The staff's request impacted all plants using surveillance data for their weld material evaluations, regardless of whether the data was from the plant's own surveillance program, an integrated surveillance program (in accordance with 10 CFR Part 50, Appendix H), or the surveillance program from another plant. The NRC staff provided an overview on use of surveillance data and the ratio procedure in public meetings held in November 1997 and February 1998. The overview also provided several examples of specific situations that could occur when applying the surveillance data to the PTS and P-T limit evaluations.

In addition to making adjustments to the data in accordance with the ratio procedure, adjustments may be required to account for differences in the irradiation environment, and specifically for the difference in the irradiation temperatures for the surveillance capsule and the RPV inner wall. This temperature adjustment reflects observations that, within the temperature range for operation of U.S. nuclear power plants, a higher irradiation temperature results in reduced embrittlement, while a lower irradiation temperature results in higher embrittlement. This adjustment is made using the down-comer or cold-leg temperature as the reference temperature for the vessel wall and the surveillance capsules. The difference in the temperatures for the surveillance capsule and inner vessel wall is accommodated using a degree-per-degree approach, wherein the surveillance data for a surveillance capsule irradiated at a temperature of X° above or below the vessel wall operating temperature would have X° added to or subtracted from each measured value of ΔRT_{NDT}, respectively.

As indicated in Regulatory Position 2.1 of RG 1.99, Revision 2, two or more credible surveillance data sets may be used to determine the ART values of the RPV beltline materials. Surveillance data are deemed credible in accordance with criteria described in the Discussion section of RG 1.99, Revision 2, and in 10 CFR 50.61(c)(2)(i).

10 CFR 50.61 and Regulatory Position 2.1 of RG 1.99, Revision 2, specify the process to be used to determine the ART with the availability of credible surveillance data. This position specifies that if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, the measured values of ΔRT_{NDT} should be adjusted by multiplying them by the ratio of the chemistry factor (CF) for the vessel weld, as determined from Table 1 of the RG, to that for the surveillance weld, as determined from the same Table. This is indicated below in mathematical form by Equation 2-1:

$$\text{Ratio Adjusted } \Delta RT_{NDT} = \frac{\text{Table } CF_{\text{Vessel Chem.}}}{\text{Table } CF_{\text{Surv. Chem.}}} \times \text{Measured } \Delta RT_{NDT} \qquad (2\text{-}1)$$

where Table $CF_{\text{Vessel Chem.}}$ represents the CF for the *vessel weld*, as determined from Table 1 of the RG and based on its copper and nickel contents, and Table $CF_{\text{Surv. Chem.}}$ represents the CF for the *surveillance weld*, as determined from the same Table and based on its copper and nickel contents. The presumption made in this Regulatory Position is that a commonality exists in the weld wire heat number for the vessel and the surveillance welds.

Regulatory Position 2.1 of the RG states the need for making adjustments to the measured ΔRT_{NDT} surveillance data because there may be considerable variability in the copper chemistry contents when multiple welds are fabricated from a given material heat. The variability in the copper contents may be especially significant if the welds are made from copper-coated weld wire, where the thickness of copper coating may vary along the length of the wire spool. In contrast, except for welds fabricated by CE using an additional nickel wire feed, the nickel contents for welds fabricated from a given material heat do not normally vary as significantly as do the copper contents. Regulatory Position 2.1 does not specify any need for similar adjustments to the measured ΔRT_{NDT} data for base metals (i.e., plate and forging materials), because base metals generally have more homogeneous alloying (chemical) contents than do welds.

According to methods of Position 2.1 of RG 1.99, Revision 2, an interim CF value based on the surveillance data is determined from the measured ΔRT_{NDT} (base metals) or the ratio-adjusted ΔRT_{NDT} (weld metals, from Equation 2-1) according to Equation 2-2:

$$CF_{Surv} = \frac{\sum_{i=1}^{n} [A_i \times f_i^{(0.28 - 0.10 \times \log fi)}]}{\sum_{i=1}^{n} [f_i^{(0.56 - 0.20 \times \log fi)}]} \qquad (2\text{-}2)$$

where n is the number of surveillance data points, A_i is the measured ΔRT_{NDT} (base metals) or the ratio-adjusted ΔRT_{NDT} (weld metals), and f_i is the fluence for the ith surveillance data point. The credibility of the CF_{Surv} is determined by comparing the residuals of the measured ΔRT_{NDT} data to the predicted ΔRT_{NDT} data (as calculated using the CF_{Surv}). If these residuals meet the credibility criteria of the RG, then the CF_{Surv} becomes the CF for determining the ΔRT_{NDT} value for the RPV beltline material, and the value of σ_Δ used in the calculation of the margin term may be reduced by half.

Five example cases concerning surveillance data were described in the public meetings in November 1997 and February 1998. The processes for evaluating the surveillance data in these cases are described below:

Case 1: Credible surveillance data (weld material) only from the plant to be evaluated:

a. Determine the interim CF, CF_{Surv}, using as-measured ΔRT_{NDT} data (Equation 2-2);
b. Evaluate the credibility of CF_{Surv} (compare measured ΔRT_{NDT} to predicted ΔRT_{NDT} using CF_{Surv} and check that the differences are less than σ_Δ) — found to be credible in this case;
c. Perform the ratio procedure adjustment to the measured ΔRT_{NDT} data (to match the RPV weld chemistry);
d. Determine that no irradiation environment (temperature) adjustments are required;
e. Reevaluate CF_{Surv} from Equation 2-2;
f. Use CF_{Surv} and reduce σ_Δ by half to evaluate status of RPV integrity per Position 2.1 of RG 1.99, Revision 2.

Case 2: Noncredible surveillance data (base metal) from the plant to be evaluated and Table CF is conservative:

a. Determine the interim CF, CF_{Surv}, using as-measured ΔRT_{NDT} data (Equation 2-2);
b. Evaluate the credibility of CF_{Surv} — found to be noncredible in this case, as the differences between the measured ΔRT_{NDT} and predicted ΔRT_{NDT} using CF_{Surv} exceed σ_Δ;
c. Determine that no chemical composition or irradiation environment (temperature) adjustments required;
d. Compare the CF_{Surv} from Equation 2-2 to the Table CF — Table CF is higher and, therefore, conservative;
e. Use the Table CF and full value of σ_Δ to evaluate status of RPV integrity, since the surveillance data are noncredible and the Table CF is conservative.

Case 3: *Noncredible surveillance data (base metal) from the plant to be evaluated and Table CF is nonconservative*:

a. Determine the interim CF, CF_{Surv}, using as-measured ΔRT_{NDT} data (Equation 2-2);
b. Evaluate the credibility of CF_{Surv} — found to be noncredible in this case as differences between measured ΔRT_{NDT} and ΔRT_{NDT} predicted using CF_{Surv} exceed σ_Δ;

c. Determine whether Table CF is conservative by comparing measured ΔRT_{NDT} to ΔRT_{NDT} calculated using Table CF and check if differences are less than 2 σ_Δ — in this case, the Table CF is nonconservative, as some differences between measured and calculated ΔRT_{NDT} exceed 2 σ_Δ;
d. Determine that no chemical composition or irradiation environment (temperature) adjustments are required;
e. Use the CF_{Surv} and full value of σ_Δ to evaluate status of RPV integrity, since the surveillance data are noncredible and the Table CF is nonconservative.

Case 4: Surveillance data (weld metal) from the plant to be evaluated and from other plants (different nuclear steam supply system or NSSS vendors):

a. Determine the interim CF, CF_{Surv}, using as-measured ΔRT_{NDT} data only from the plant to be evaluated (Equation 2-2), since the in-vessel plant surveillance data do not require a temperature adjustment;
b. Evaluate the credibility of CF_{Surv} — found to be credible. Surveillance data from other plants were not used because they require temperature adjustment, while the in-plant surveillance data are credible;
c. Perform the ratio procedure adjustment to the measured ΔRT_{NDT} data (to match the RPV weld chemistry);
d. Determine that no irradiation environment (temperature) adjustments required;
e. Reevaluate the CF_{Surv} as determined from Equation 2-2;
f. Use CF_{Surv} and reduce σ_Δ by half to evaluate status of RPV integrity per Position 2.1 of RG 1.99, Revision 2.

Case 5: Surveillance data (weld metal) only from plants other than the plant to be evaluated (different NSSS vendors):

a. Start with the surveillance data from plant(s) with the same NSSS vendor as the plant to be evaluated, since capsules from a plant with the same NSSS vendor operate in a similar nuclear environment;
b. Evaluate the interim CF, CF_{Surv} (Equation 2-2) using as-measured ΔRT_{NDT} data identified in Step a—if the data are from multiple heats or plants, then chemical composition and irradiation environment (temperature) adjustments (to the average chemical composition and irradiation temperature of the surveillance data) may be required before evaluating CF_{Surv};
c. Evaluate credibility of CF_{Surv} —found to be credible in this case. Surveillance data from other plants with different NSSS vendor were not used because they require temperature and other adjustments, while surveillance data from the same NSSS vendor are credible.
d. Adjust the measured ΔRT_{NDT} data in accordance with the ratio procedure to match the RPV weld chemistry and adjust the irradiation temperature of the plant to be evaluated;
e. Reevaluate the CF_{Surv} as determined from Equation 2-2 using the adjusted ΔRT_{NDT} values from Step d, above;

f. Use CF_{Surv} and reduce σ_{Δ} by half to evaluate status of RPV integrity per Position 2.1 of RG 1.99, Revision 2.

These examples cases are an indication of the possible situations that may be encountered when applying plant-specific surveillance data to the structural integrity assessments for a plant's RPV. The staff recognizes that situations may arise where alternative methods of evaluating the surveillance data may be necessary. The staff will review any alternative methods of evaluating the surveillance data on a case-by-case basis.

In BAW–2325, the B&WOG evaluated all of the available surveillance data for the weld metals in the B&WOG Reactor Vessel Working Group plants, which are plants with Westinghouse and B&W NSSS designs having B&W fabricated vessels. These welds were fabricated with the automatic submerged-arc process using copper-plated manganese-molybdenum-nickel (Mn-Mo-Ni) filler wire and Linde 80 flux.[3] As a part of the B&WOG evaluation of the surveillance data for Linde 80 welds, all of the Charpy-V notch data were reevaluated using a hyperbolic tangent curve fitting program to achieve consistency in the interpretation of the data. In response to NRC questions, the B&WOG assembled Supplement 1 to BAW–2325, providing both the original interpretation of the surveillance data (generally from hand-fits to the data) and the hyperbolic tangent curve fitting interpretation. In the report, the B&WOG documented its systematic evaluation of the surveillance data for each weld wire heat number, after considering many of the concepts described previously in the example cases. Revision 1 to BAW–2325 provides a final analysis of the data for each weld wire heat, after considering all corrections to the fitted surveillance data as well as appropriate adjustments to capsule fluences.

2.4 Requests for Additional Information

2.4.1 Contents of the Requests for Additional Information

After reviewing the data collected by the owners groups, the staff issued a number of requests for information (RAIs) to certain pressurized water reactor (PWR) and BWR plants. These RAIs were issued from March to August 1998. The RAIs included blank tables for licensees to update the alloying chemistries of the beltline welds (or the limiting plate material if applicable), and to assess its surveillance data after reviewing the appropriate owners group topical report. The RAIs also asked the licensees to determine how the changes to the best estimate chemistry values would impact the structural integrity assessments for their facility relative to the requirements of 10 CFR 50.60; 10 CFR Part 50, Appendices G and H; and 10 CFR 50.61.

2.4.2 Status of the Staff's Review of the Responses to the RAIs

All plants that received RAIs submitted their responses. The updated data and references submitted by licensees as a part of the GL 92–01, Revision 1, Supplement 1, review were entered into the newly-developed Microsoft Access® version of the Reactor Vessel Integrity Database (RVID). The RVID is discussed further in Chapter 6 of this report.

(3) The identifying characteristic of these welds is the heat identification of the weld wire used for fabrication of the welds. The heat identification indicates the melt of material from which the wire was fabricated. All lengths (coils) of wire from the same melt material have the same heat identification number. Since many coils can be made from a single melt, the same heat identification number can be associated with the fabrication of multiple vessel and surveillance welds.

As a result of its review of the responses to the RAIs on GL 92–01, Revision 1, Supplement 1, the staff identified a number of issues with the manner in which some of the licensees were applying the updated surveillance capsule and chemistry data to their PTS and/or P-T limit assessments. The issues are:

(1) Several units have surveillance data where one or more data points do not meet the credibility criteria in RG 1.99, Revision 2, and 10 CFR 50.61 (the pressurized thermal shock rule). In some cases, licensees from the subject units used noncredible surveillance data to calculate the CFs, and a reduced margin term for calculating RT_{PTS}. The staff's RAIs on GL 92–01, Revision 1, Supplement 1, contained the following statement:

"....10 CFR 50.61(c)(2) specifies that licensees will consider plant-specific information (e.g., operating temperature and surveillance data) to verify that the RT_{NDT} for each vessel beltline material is a bounding value. Regulatory Guide 1.99, Revision 2, describes two methods for determining the amount of margin and the chemistry factor used in determining RT_{NDT}. If the evaluation of the surveillance data indicate that the surveillance data set is not credible and the measured values of ΔRT_{NDT} are less than the projected mean from the Tables plus the generic $2\sigma_{\Delta}$, the chemistry factor may be calculated using either Position 1.1 or Position 2.1; however, the full margin term must be applied.

The method chosen must bound all the surveillance data to be in compliance with 10 CFR 50.61(c)(2)."

Section 2.3 of this report presented five example cases as an indication of the typical situations that may arise when applying plant-specific surveillance data to the CF calculations. The staff used, where applicable, the full value of σ_{Δ} in this review when non-credible surveillance data were used to calculate the CF.

(2) Two licensees with CE fabricated RPVs proposed to use their plant-specific chemistry data in their RPV integrity calculations; however, these data are nonconservative when compared with the CE topical reports. The staff used the data from the CE topical reports during the review.

(3) Some licensees submitted fluence evaluations in topical reports that were reviewed as part of this effort. The submittal from one unit did not justify the proposed reduction in fluence. Although the current docketed fluence value was maintained in the RVID, the staff recommended that the licensee use calculational methods to verify the EOL fluence value when the next surveillance capsule is removed from the vessel and analyzed.

(4) The response to the RAI on GL 92–01, Revision 1, Supplement 1, for a multi-unit nuclear station did not reflect the fluence, $RT_{NDT(U)}$, and σ_U values that were updated in its recent pressure-temperature (P-T) limits submittal. The staff used the information from the P-T limits submittal in the RPV integrity calculations.

(5) One licensee was in dispute with the CEOG regarding the identity of the surveillance weld for its plant. Although this discrepancy did not impact the current P-T limits for the plant, it did affect the chemical composition of the subject weld reported by the CEOG. The licensee responded to the staff's closeout letter and indicated that the identity of the surveillance weld had been investigated and reconciled with the CEOG reports. Therefore this issue has been resolved.

(6) A licensee submitted a revised $RT_{NDT(U)}$ value for a beltline weld in one of its RPVs; however, the licensee did not provide the basis for the change in the submittal. The staff informed the licensee that it would use the previous docketed plant-specific $RT_{NDT(U)}$ value. Therefore, the staff did not implement the change to the $RT_{NDT(U)}$ value when updating the RVID.

(7) The shell courses of one licensee's RPV have differing thicknesses. The staff had previously contacted the licensee regarding this issue and reiterated the discrepancy during the GL review effort. This issue was documented in the updates to the RVID as an aid for reviewing any proposed changes to the P-T limit curves currently in the plant's Technical Specifications.

The staff identified these issues in the appropriate RVID reference sections and summary sheets during its update of the database. The staff also informed each licensee associated with these issues of its basis for making a change to the docketed data and inputting the amended data into the updated database.

2.4.3 Closeout of Generic Letter 92–01, Revision 1, Supplement 1

Each licensee received a letter to close out GL 92–01, Revision 1, Supplement 1. These letters were issued to licensees in 1999. The letters recommended that licensees review the RPV integrity data for their plants provided in the revised version of RVID. Licensees may do so by downloading the RVID installation diskettes from a web site that is linked to the NRC homepage (http://www.nrc.gov/NRR/RVID/index.html). It should be noted that the website for the original version of the RVID was not linked to the NRC homepage. The closeout letters also informed licensees that the staff would assume that data entered into the RVID are acceptable if no comments were received by September 1, 1999. The closeout letters concluded by stating that future submittals on P-T limits, pressurized thermal shock (PTS), or upper shelf energy (USE) should reference the most current RPV integrity information.

3 SIGNIFICANT REGULATORY AND RESEARCH ACTIVITIES ON RPV INTEGRITY

3.1 Background

Since the issuance of NUREG–1511, Supplement 1 (Ref. 20), in October 1996, several developments have occurred that are expected to significantly affect the technical assessment of RPV integrity issues. This chapter will discuss these developments under two broad categories: (1) research activities to advance the understanding of RPV integrity parameters; and (2) consensus Codes and Standards activities to formalize alternatives to the established RPV integrity assessment methodologies.

3.2 Research Activities

Four topics will be discussed in the area of research activities that have advanced the understanding of RPV integrity parameters. The first is the continuing work by NRC staff, NRC contractors, and industry representatives (under the auspices of American Society for Testing and Materials (ASTM) Committee E10) on the development of new embrittlement correlations for RPV materials. The NRC's Office of Nuclear Regulatory Research (RES) began this work in 1993 in conjunction with efforts by ASTM Committee E–10 in order to determine whether the existing database of commercial power reactor surveillance data (which currently consists of about 720 surveillance data points) could be used to improve upon the embrittlement correlations stated in US NRC Regulatory Guide (RG) 1.99, Revision 2 (which were established based on a database of about 177 data points). This work may provide the basis for revising both RG 1.99 and ASTM Standard E900 (Ref. 32). As part of the effort to revise these documents, the work on the new embrittlement correlations has also expanded to include an assessment of the margin or uncertainty to be applied when using the new correlations and to develop a methodology for the assessment and use of plant-specific surveillance data.

The initial analysis of the data conducted by an NRC contractor has been documented in Topical Report NUREG/CR–6551 (Ref. 33). Industry representatives on ASTM Committee E10 have supported this work by providing additional quality assurance reviews of the surveillance data in the NRC's Power Reactor Embrittlement Data Base and correcting or filling in missing data. The ASTM committee members have also reviewed the models developed in NUREG/CR–6551, and questions raised within the committee have led to the need for some reanalysis of the NUREG/CR–6551 models. Work on refining the models, completing the uncertainty or margins analyses, and assessing the surveillance data is expected to continue at least through fall 2000.

The second topic has been the research performed to develop new RPV flaw distributions. Probabilistic fracture mechanics (PFM) evaluations of reactor pressure vessel integrity require characteristics of assumed flaws as one of the key input parameters. In lieu of deterministic applications of a fixed flaw size and geometry, flaw density and flaw size distributions are used to characterize a population of flaws for use in the PFM evaluations. For many years, the "Marshall distribution" (Ref. 34) has provided the basis for PFM calculations. Two recent initiatives by RES are focused on improving existing flaw density and flaw size distribution data. In the first initiative, researchers at the Pacific Northwest National Laboratories (PNNL) have been using state-of-the-art ultrasonic testing (UT) inspection equipment and evaluation tools to inspect the welds in vessels from decommissioned nuclear plants. These nondestructive evaluations have been accompanied by destructive confirmation of the indications identified by the UT inspections. Preliminary results of this work are provided in References 35 and 36.

The second initiative refers to an expert judgement process that is used to classically resolve specific technical issues for which there is scientific uncertainty. In this case, the expert judgement process has been employed to review, interpret, and supplement available information on RPV fabrication processes and RPV flaw density and flaw size distributions. RES expects to issue a NUREG report on this work by the end of 2001.

The third topic actually transcends both of the broad categories laid out at the beginning of this chapter, that is development of the Master Curve methodology. The Master Curve data methodology relies on the similarity of the fracture toughness-temperature transition behavior for ferritic steels, and provides a framework for enabling structural integrity assessments based on limited fracture toughness data sets from small test specimens (e.g., precracked Charpy-V specimens). This technology was originally developed at Valtion Teknillinen Tutimuskeskus (VTT, a Finnish Research Laboratory) (Ref. 37), and substantial funding for research in the United States has been provided by NRC's RES. A significant amount of additional research into the Master Curve methodology has also been supported by WOG, BWOG, CEOG, and EPRI.

The final topic under research activities is a program initiated by RES to reevaluate the PTS screening criteria in 10 CFR 50.61. This program will combine the information from previous NRC work in the 1980s (i.e., the Integrated Pressurized Thermal Shock program) with improved flaw distributions, new embrittlement correlations, and a better understanding of topics such as RPV weld chemistry variability. Issues regarding the identification of potential PTS transients, thermo-hydraulic calculations, and the development of acceptable risk metrics will also be addressed. This program has also been developed to include substantial participation by the industry and the public throughout the process.

3.3 Codes and Standards Activities

Several actions have also been taken by consensus Codes and Standards bodies to provide a framework for use of the Master Curve methodology. ASTM Standard E1921 (Ref. 38) was passed in 1997 and provides methods for conducting Master Curve testing and data analysis. Subsequently, ASME Code Case N–629 (Ref. 11), which was approved in 1998, defines a relationship between the Master Curve test parameter (T_0, a temperature defined to correspond to a fracture toughness of 100 MPa$\sqrt{m}$) and the ASME Code methodology for indexing fracture toughness properties to a reference temperature. In this Code Case, a reference temperature, RT_{T0}, as given by $RT_{T0} \equiv T_0 + 35°F$, is provided as an alternative for indexing temperature to RT_{NDT}. During summer 1999, one licensee (Wisconsin Public Service Corporation, the licensee for the Kewaunee Plant) submitted a license amendment and requested an exemption for NRC approval to apply Code Case N–629 and the Master Curve methodology to the analysis of their limiting RPV circumferential beltline weld (Ref. 39). The NRC's review of this submittal is expected to be completed by the end of Year 2000.

In the area of consensus Codes and Standards activities, two additional ASME Code Cases have been passed that are of interest to RPV integrity evaluations. ASME Code Case N–588 (Ref. 10) was developed to permit licensees to postulate a circumferentially-oriented flaw when evaluating a circumferential weld for RPV P-T limits. Previously, in the latest edition (1995, Edition through Summer 1996 Addenda) of Appendix G to the ASME Code, which has been invoked by reference in 10 CFR 50, Appendix G, licensees were required to postulate axially-oriented flaws in all RPV beltline materials when evaluating RPV P-T limits. Based on the metallurgical processes associated with welding, large axially-oriented flaws perpendicular to a circumferential weld seam are extremely unlikely. In fact, the ASME Code reference flaw (¼ wall thickness in depth with a 6:1 aspect ratio) would extend not only across the width of a nominal RPV circumferential shell weld but into the base material beyond, making the axial flaw assumption even more non-physical for circumferential shell welds.

The other ASME code case of interest is N-640 (Ref. 9, formerly listed as N-626), which permits the use of the lower bound static initiation fracture toughness curve (K_{Ic}) instead of the lower bound crack arrest fracture toughness curve (K_{Ia}) for developing RPV P-T limits. The K_{Ia} fracture toughness curve, being based on dynamic and crack arrest data, has provided a conservative basis in Appendix G to the ASME Code for establishing P-T limits. With the approval of this Code Case, the ASME Code concluded that the use of the K_{Ic} fracture toughness curve is technically justified, since the intent of the ASME Code procedures and the NRC regulations has been based on preventing crack initiation.

10 CFR Part 50, Appendix G, requires that the P-T limits for nuclear power generation facilities must be at least as conservative as those that would generated if Appendix G to the ASME Code were used to establish the curves. Since the methods of Code Cases N-640 and N-588 will generate P-T limit curves that are less conservative than those that would generated using Appendix G to the ASME Code, licensees must request exemptions under the regulatory framework of 10 CFR 50.60(b) and 10 CFR 50.12 if they desire to generate their P-T limit curves using the Code Case methods. The staff will review applications for exemptions to use Code Case N-640 or Code Case N-588 on a case-by-case basis.

4 RECENT INDUSTRY EFFORTS REGARDING BWR RPV WELD INSPECTION REDUCTION (BWRVIP-05)

4.1 Overview

In 1995 the BWR Vessel and Internals Project (BWRVIP), a special industry technical review group formed by the BWROG to focus on resolution of reactor vessel and internals degradation issues, submitted a proprietary report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP–05)" (Ref. 13), for staff review and approval. The BWRVIP–05 report evaluated the current inspection requirements for the RPV shell welds in BWRs, formulated recommendations for alternatives to the inspection requirements, and provided a technical basis for the recommended alternatives. In the report, and as revised in subsequent submittals, the BWRVIP proposed to reduce the scope for augmented inspections of the circumferential RPV shell welds from "essentially 100 percent" of the welds (i.e., as defined in 10 CFR 50.55a(g)(6)(ii)(A)(*2*)) to essentially zero percent of the welds, except for the portions of the welds located at intersections of the axial and circumferential welds, where approximately 2-3 percent of the circumferential welds would be inspected. Revised criteria for the performance of successive and additional inspections were also recommended.

4.2 Background

On May 12, 1997 (Ref. 40), the BWRVIP and NRC staff briefed the Commission on issues related to the requirements for a full inspection of reactor pressure vessel shell welds. The transcript of the Commission meeting of May 12, 1997, the Commission's Staff Requirements Memorandum (Ref. 41), and the meeting summaries related to this issue are available in the Commission's Public Document Room, 2120 L Street NW., Washington, DC 20555.

On August 7, 1997, the staff issued Information Notice (IN) 97–63, "Status of NRC Staff's Review of BWRVIP-05" (Ref. 42), regarding licensee requests for relief from the augmented inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A). In the information notice, the staff stated that it would consider technically-justified requests for reliefs from the augmented examination requirements in accordance with the provisions for alternative programs stated in sections (a)(3)(i)and(ii) and (g)(6)(ii)(A)(*5*) to 10 CFR 50.55a. Acceptably-justified requests were to be considered from BWR licensees who were scheduled to perform inspections of the RPV shell welds during the fall 1997 or spring 1998 outage seasons. In the information notice, the staff stated it would consider inspection delays of up to two operating cycles for the RPV circumferential shell welds only. The staff also stated that licensees will still need to perform the required augmented inspections of "essentially 100 percent" of all axial RPV shell welds.

The acceptability of such requests was based on plant-specific information submitted by the licensee. The staff granted schedular reliefs to defer the inspections of the RPV circumferential shell welds for four BWR units scheduled to enter refueling outages (RFOs) during the fall 1997 outage season, and for two BWR units scheduled to enter RFOs during the spring 1998 outage season.

On August 14, 1997, the staff forwarded to BWRVIP its independent safety assessment (ISA) of the BWRVIP–05 document (Ref. 43). The staff's ISA was a multi-disciplinary, risk-informed review of the safety implications of reducing the inspection scope for the RPV circumferential shell welds as proposed in the BWRVIP–05 report. It provided a description of the two degradation mechanisms (fatigue and stress corrosion) that have the potential to initiate RPV cracking or to promote the growth of existing flaws, and of the limiting transients of concern. Also transmitted with the ISA was additional guidance on the type of information the staff would need to assess plant-specific requests for relief from the ISI requirements of 10 CFR 50.55a(g)(6)(ii)(A).

Further work was performed by both the staff and the industry to assess more fully the risk associated with beyond-design-basis events for both the axial and circumferential welds at fluence levels projected to be reached later in life at some plants. This additional work included: (1) studies of potential precursor events to better quantify the potential for cold overpressure events in BWRs, (2) additional probabilistic fracture mechanics analyses both to understand the sensitivities to various parameters and to support an uncertainty analysis, and (3) an assessment of the proposed changes in inspection requirements relative to the probability of vessel failure.

On May 7, 1998, the staff issued IN 97–63, Supplement 1 (Ref. 44), which informed BWR licensees that the staff was extending the applicable periods to the fall 1998 and spring 1999 outage seasons.

The staff concluded in its ISA that beyond-design-basis events occurring during plant shutdown (e.g., injection of cold water into the RPV at pressure or excessive pressurization of the cold vessel) could lead to cold overpressurization events that could challenge vessel integrity. Specifically, the staff identified a transient at a foreign BWR of U.S. design, in which the RPV was subjected to high pressure (7.9 MPa or 1150 psig) while at a low temperature (26°C–31°C or 79°F–88°F). This cold overpressure transient was not included as a design basis event for BWRs and was not considered in the BWRVIP–05 report, which focused only on design-basis events. However, the recognition of this transient led the staff to conclude that cold overpressure transients are safety significant and need to be considered. Accounting for these precursor and actual events, the staff estimated a frequency of cold overpressurization events that could challenge the RPV integrity at cold shutdown.

The industry's response to the staff's ISA concluded that condensate and control rod drive (CRD) pumps could cause conditions that could lead to cold overpressure events that could challenge vessel integrity. Specifically, the industry gave the following justifications as the bases for concluding that most of the other BWR injection or spray systems would not contribute to the overall frequency for the occurrence of cold overpressurization events:

1. The shutoff head to low pressure coolant spray or injection systems is low (e.g., shutoff heads to the LPCS/LPCI pumps), so that the RPV remains within the acceptable limits of the pressure-temperature (P-T) curves even at shutdown temperatures (i.e., these systems can be activated only under low pressure).

2. Overfilling and pressurization to the shutoff head for high pressure coolant spray (HPCS) systems is very unlikely because the system automatically trips on the high-water level indications.

3. The reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems are steam driven and are not in use during cold shutdown conditions.

4. Operation of the standby liquid control (SLC) system requires a series of deliberate operator actions (e.g., manual pump activation) and is unlikely to occur without adequate monitoring.

Thus, RCIC, HPCI, HPCS, LPCI, LPCS, and the SLC systems were considered to have a negligible impact on the frequency for a cold overpressurization event. The BWRVIP's estimate of the frequency of overpressurization events that could challenge the RPV was 9.5×10^{-4}/yr for BWR-4 designs and 9×10^{-4}/yr for BWR designs other than BWR-4. This frequency estimate is comprised of 1.5×10^{-4}/yr from condensate injection, 7×10^{-4}/yr from CRD injection (vessel pressure testing), and 1×10^{-4}/yr from loss of RWCU (3×10^{-5}/yr from loss of RWCU in BWR designs other than BWR-4). The staff considered the frequency contributions from these systems to be reasonable.

The staff noted, however, that the BWRVIP concluded that the potential contribution from other injection sources (i.e., RCIC, HPCI, HPCI, HPCS, LPCI, LPCS, and SLC systems) would have a negligible impact on the frequency for the occurrence of a cold overpressurization event. However, because historical data indicated that actual inadvertent injections of these systems have occurred, the staff did not consider their contribution to the cold overpressurization frequency to be negligible. Therefore, the staff concluded that the NRC-estimated frequency for cold overpressurization resulting from inadvertent injections of these systems (6×10^{-5}/yr) should be added to BWRVIP's estimate of about 9×10^{-4}/yr. The staff therefore estimated the total frequency for cold overpressurization to be 1×10^{-3}/yr.

4.3 NRC Assessment of BWRVIP–05

To estimate the conditional failure probability (P(FIE)) for the BWR RPV vessel, the staff performed independent generic and plant-specific probabilistic fracture mechanics (PFM) analyses using the FAVOR Code. The FAVOR Code, which was developed by the Oak Ridge National Laboratory (ORNL), performs millions of random deterministic vessel simulations to determine the mean P(FIE) for a vessel subjected to a specific transient. To perform the independent PFM analyses, the staff used the following data as inputs to the simulations: (1) the operational data from the referenced foreign transient, (2) updated RPV flaw density and size distributions based on data developed at the NRC's Pressure Vessel Research Users Facility (PVRUF), and (3) the material chemistries and fluences developed by different vessel fabricators for the generic RPV evaluations and the material chemistries and fluences of bounding RPVs for the plant-specific evaluations.

The BWRVIP failure frequency for the limiting circumferential welds was 9.0×10^{-10} /yr ((9.0×10^{-4}/yr event frequency for a BWR–3) x (1.0×10^{-6} conditional probability of failure)). In contrast, the staff determined the limiting plant-specific failure frequency for circumferential welds at 32 effective full power years to be 8.2×10^{-8}/yr ((1×10^{-3}/yr event frequency) x (8.2×10^{-5} conditional probability of failure)). As depicted in NUREG–1560, Vol. I (Ref. 45), core damage frequencies (CDF) for BWR plants were reported to be approximately 10^{-7}/yr to 10^{-4}/yr. In addition, Regulatory Guide (RG) 1.154 (Ref. 46) indicates that PWR plants are acceptable for operation if the plant-specific analyses predict that the mean frequency of through-wall crack penetration for pressurized thermal shock events is less than 5×10^{-6}/yr. Since the failure frequencies for circumferential welds in BWR RPVs were significantly lower than the failure probabilities specified in RG 1.154 and NUREG-1560, Vol. I, and since additional volumetric inspections would not significantly improve upon the already acceptably low failure probabilities for circumferential welds, the staff concluded that the proposal to eliminate the ISI requirements for RPV circumferential welds was justified. Therefore, in its final safety evaluation on BWRVIP-05 (Ref. 14), the staff informed the BWRVIP that the proposal to eliminate BWR vessel circumferential weld examinations is acceptable.

On November 11, 1998, the staff issued GL 98–05 (Ref. 47) to inform licensees of BWRs that they may request permanent relief from meeting the inservice inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A) for circumferential RPV shell welds during the remainder of the current 40-year license terms for their facilities. In the GL, the staff stated that licensees could justify using the BWRVIP-05 report as the basis for reducing the inspection scope by (1) demonstrating that the circumferential RPV shell welds would continue to satisfy the limiting conditional failure probability for circumferential welds at EOL as cited in the staff's safety evaluation of July 28, 1998; and (2) demonstrating that they will have implemented operator training and established operating procedures to ensure that the probability for the occurrence of a cold overpressurization event would be limited to the frequency specified in the staff's safety evaluation of July 28, 1998. In the GL the staff also informed the licensees that they would still need to perform the required inspections of "essentially 100 percent" of all axial RPV shell welds.

Unlike circumferential flaws, the failure frequencies estimated by the staff (Ref. 14) for BWR axial welds were relatively high, about 4.4×10^{-4}/yr. Conservatisms in the analyses, such as using the limiting material properties and chemistry for the inside surface flaws in the axial welds at the location of peak end-of-license (EOL) azimuthal fluence, excluded the RPV axial welds as a near-term safety concern. Nonetheless, to resolve the long-term potential safety concern, the staff pursued this subject with the BWRVIP to ensure that the estimated failure frequency for BWR axial welds is significantly lower than indicated by the staff's estimate for axial RPV shell welds. In May 1999, the NRC staff met with the BWRVIP to discuss revised assumptions and criteria for the axial weld failure probability analyses. The BWRVIP submitted a revised assessment for BWR axial welds on November 12, 1999 (Ref. 48). By letter dated March 7, 2000 (Ref. 49), the staff issued its evaluation of the revised assessment. In this letter, the staff concluded that the RPV failure frequencies for the limiting axial welds in the BWR fleet are below the applicable regulatory limits, given the assumptions used in the evaluation. However, the staff also noted that this conclusion applied only to initial 40-year license periods, and that consideration of BWR vessel welds for license renewal terms would require plant-specific assessments by the license renewal applicants.

In the letter of March 7, 2000, the staff also noted that an expert judgement process has been employed to review, interpret, and supplement available information on RPV fabrication processes and RPV flaw density and flaw size distributions. The staff concluded that, should the results of the expert judgement process prove to be more conservative than the assumptions used in the BWRVIP's assessment (Ref. 48), the BWRVIP would be required to re-evaluate BWR axial welds using the results of the expert judgement process.

5 REACTOR PRESSURE VESSEL THERMAL ANNEALING

5.1 Thermal Annealing Regulation

Thermal annealing is a process which is used to mitigate the effects of radiation embrittlement on RPV materials. The NRC has established a regulation, 10 CFR 50.66 (Ref. 16), to provide the regulatory framework for the thermal annealing of RPVs. The regulation addresses the critical engineering and metallurgical aspects of thermal annealing and requires the following actions: (1) submittal of an engineering plan and analysis for thermal annealing in a thermal annealing report (TAR); (2) submittal of a report detailing the estimated degree of fracture toughness recovery that will be accomplished before implementing the annealing process; (3) submittal of a report confirming that the thermal annealing process was performed in accordance with the TAR; and (4) public meetings to be held both before and after the process is implemented to allow interested parties to make inquiries.

5.2 Regulatory Guides and Technical Codes and Standards for Thermal Annealing

5.2.1 Regulatory Guide 1.162

RG 1.162 (Ref. 17) was issued by the NRC to provide guidance for the thermal annealing of RPVs. The RG contains a detailed listing of metallurgical and engineering issues that should be addressed in an application to implement a thermal annealing process. RG 1.162 presents three acceptable methods for estimating the degree of fracture toughness recovery resulting from a thermal annealing process:

- testing of RPV surveillance program materials;
- removal of specimens from the RPV beltline;
- a generic computational method.

5.2.2 ASTM Standard Procedure E-509

General guidance for in-service annealing is given in ASTM Standard E509-86 (Ref. 50). Specifically, ASTM Standard E509-86 prescribes general procedures for conducting an inservice thermal annealing of a RPV and for demonstrating its effectiveness and the degree of recovery in fracture toughness.

5.2.3 ASME Code Case N-557 on Thermal Annealing

ASME Code Case N-557 (Ref. 51) was developed in 1995 by a special ASME task group to provide guidance specifically focused on the structural engineering aspects of thermal annealing. The Code Case addresses how annealing conditions (temperature and duration), temperature monitoring, evaluation of loadings, and nondestructive examination will be applied to the thermal annealing process. Code Case N-557 was formally approved by ASME in March 1996.

5.3 Palisades Thermal Annealing Report

In October 1995, CPCo (currently named Consumers Energy), the licensee for the Palisades plant, submitted a TAR for annealing of the RPV at the Palisades Nuclear Plant (Ref. 52). CPCo submitted the TAR after determining that the materials in the vessel might not satisfy the screening criteria of the PTS rule (10 CFR 50.61) through the end of license (EOL) for the unit. CPCo projected that an annealing treatment of the Palisades RPV would result in recovery of 80 to 90 percent of the fracture toughness loss due to neutron embrittlement.

The staff completed a preliminary review of the Palisades TAR and requested additional information from CPCo. CPCo responded to the staff's request via a letter dated August 26, 1996 (Ref. 53). In April 1997, CPCo withdrew the TAR from the docket (Ref. 54), and instead elected to pursue other means of addressing the PTS issue for the Palisades RPV.

On April 4, 1996, CPCo submitted revised neutron fluence estimates and PTS calculations for the Palisades RPV materials (Ref. 55). CPCo's revised PTS calculations showed that the Palisades RPV would remain below the PTS screening criteria of 10 CFR 50.61 through the end of 2007. The NRC staff reviewed the CPCo analysis and agreed with some aspects of their proposed fluence reduction while continuing its review of other aspects of CPCo's fluence methodology. On December 20, 1996 (Ref. 56), the NRC staff issued an interim safety evaluation that concluded that the Palisades RPV would remain below the PTS screening criteria through the end of 2003. Subsequently, the staff determined that no additional fluence reduction could be credited based on the original CPCo analysis methodology. However, CPCo has continued to refine their neutron fluence estimates and PTS calculations in an attempt to justify continued safe operation of the Palisades RPV through the end of its original operating license (2007). Chapter 7 of this report further summarizes both CPCo's and the staff's efforts to re-evaluate the neutron fluences for the Palisades RPV.

5.4 Department of Energy/Industry Annealing Demonstration Project

The Department of Energy (DOE) and the nuclear industry have sponsored engineering projects for demonstrating the feasibility of thermal annealing in U.S. light water reactors. The contracts for two of these Annealing Demonstration Projects (ADPs) were announced on May 25, 1995. The Office of Nuclear Regulatory Research (RES) has represented the NRC's interests in these ADPs, which was conveyed to DOE in a memorandum of understanding (MOU) dated August 4, 1996, (Ref. 57). The feasibility of thermal annealing was demonstrated by a joint DOE/industry-sponsored ADP at the Marble Hill facility. This ADP employed an indirect gas-fired heating method. A second ADP using an electric resistance heating approach was planned for the Midland facility but subsequently canceled due to a lack of DOE funding. The Marble Hill and Midland facilities were decommissioned Westinghouse and B&W designed plants, respectively.

An NRC Commission briefing was held on August 27, 1996, to provide the Commission with an update of the status of the ADPs and their relevance to the Palisades plant. DOE and the consortium associated with the Marble Hill ADP provided a summary of the activities that had occurred during the ADP (May 30, 1996 to July 19, 1996). The NRC staff, led by RES, also presented their observations on the Marble Hill ADP. Specifically, the NRC staff discussed the status of the plans being prepared by CPCo for the Palisades plant and how the information gathered from the ADPs would be used to support the Palisades TAR and the NRC's review of the annealing activities. However, as noted previously, the planned annealing for the Palisades RPV was ultimately terminated.

Verification of the data and completion of a final report on the Marble Hill ADP were delayed because of the elimination of funds to support the projects beyond the end of the DOE fiscal year 1997. EPRI later provided sufficient funding to complete the Marble Hill report. EPRI published the final report on the Marble Hill ADP in March 1998 (Ref. 58).

The NRC published its final report, NUREG/CR-6552, "Marble Hill Annealing Demonstration Evaluation," in February 1998 (Ref. 59). Based on the review of the data provided to NRC following the annealing demonstration and the results of the thermal and structural analyses, it was concluded that the Marble Hill RPV was not adversely affected by the demonstration annealing cycle. Although the ADP was successful, there was a concern with the unpredicted severity of the thermal gradient in the RPV between the nozzles and the RPV flange. Therefore, in NUREG/CR–6552, the staff recommended that reliable analytical models be developed and validated to address this concern.

6 REACTOR VESSEL INTEGRITY DATABASE

6.1 Development of the Reactor Vessel Integrity Database

The Reactor Vessel Integrity Database (RVID) was developed following NRC staff review of licensee responses to GL 92–01, Revision 1 (Ref. 1). The database was designed and developed to reflect the current status of reactor pressure vessel integrity, with the data consolidated in a convenient and accessible manner. Some of the data categories represent inputs of docketed information; other data categories are representative of computed values that are not necessarily docketed. The programming logic used for calculations in the database follows the methodology in NRC RG 1.99, Revision 2 (Ref. 21).

RVID summarizes the properties of the reactor vessel beltline materials for each operating commercial nuclear power plant. For plants that ceased operation since issuance of the initial version of the RVID, the existing RPV data for those plants have been maintained within the RVID.

6.2 Availability of the RVID on the World Wide Web

The original RVID was a DOS application developed with FoxPro™ software. The RVID was updated with new data and references resulting from the staff's review of the nuclear industry's responses to GL 92–01, Revision 1, Supplement 1 (Ref. 3). The staff released the database in June 1999 as RVID Version 2 (RVID 2), a Windows 3.1 native application based on Microsoft Access 2.0™. The database runs in the higher Windows environments as well (e.g., Windows 95, Windows NT). The user is not required to have Microsoft Access™ to run the database.

In addition, comments from licensees on the June 1999 data and information were used to update the RVID, and RVID Version 2.0.1 was released in July 2000. The four (4) RVID 2 diskettes can be downloaded from the RVID 2 homepage (http://www.nrc.gov/NRR/RVID/index.html). RVID 2 is also linked to the NRC World Wide Web homepage under "nuclear reactors" and the "U.S. Commercial Nuclear Plants" heading.

6.3 Content of the RVID

In addition to the licensee responses to GL 92–01, Revision 1, Supplement 1, and GL 92–01, Revision 1, the following documents were included in the review process and development of the RVID 2 program: surveillance capsule reports; pressurized thermal shock and P-T limits submittals (as applicable); documents referenced in licensee-specific responses to GL 92–01, Revision 1 and GL 92-01, Revision 1, Supplement 1; and responses to the staff's requests for additional information (as applicable). RVID 2 has three tables for each plant: a PTS summary table for PWRs or a P-T limits summary table for BWRs, an USE summary table, and a surveillance data summary table. The surveillance data summary table is a new feature of the database. Additionally, RVID 2 includes sort and data search capabilities. The user can select a desired grouping of plants and then specify information categories to search and list.

The RVID 2 program has references and notes that document the source(s) of data and provide supplemental information. As a result of comments received from the industry, the staff revised RVID 2 to include reference fields at the component level that highlight notes specific to the particular forging, plate, or weld. RVID 2 has the capability to apply the RG 1.99, Revision 2, ratio procedure to the calculation of the chemistry factor. The database can also automatically determine the credibility of surveillance data in accordance with the criteria in RG 1.99, Revision 2, and 10 CFR 50.61 (Ref. 5).

Significant effort has been made to ensure that the RVID 2 program is "user friendly" and that the programming logic is accurate. The staff used the information in BAW–2325, Revision 1, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity" (Ref. 24), to highlight the relationship between the data in the B&W integrated surveillance program and the vessels to which they apply. Specific notes are included in the surveillance data and the component level note fields to relate the surveillance data to the applicable RPV materials.

6.4 Quality Assurance of RVID Data

The staff reviewed the docketed information for each plant, and provided paper-markups of data to designated RVID editors. The two editors used the paper-markups to update the RVID, and consulted with each reviewer when necessary. Each editor performed quality assurance (QA) checks of the data in preparation for and after the update process. In addition, the editors required each original reviewer to QA the data before it was released on the RVID web site. Subsequently, in the GL 92–01 closeout letters, the staff requested that licensees review the information, and provide any comments back to the NRC. The staff will review any comments and recommended changes received from the industry and input the data into the RVID, as appropriate.

6.5 Future Revisions to the RVID

The RVID-2 program will be updated when sufficient amounts of new surveillance data, chemistry data, or fluence evaluations warrant a new revision. Revisions to the RVID database diskettes will continue to be released on the World Wide Web for downloading. The NRC technical monitor will mail the diskettes to requesters who do not have access to the World Wide Web.

7 SIGNIFICANT PLANT-SPECIFIC REVIEWS

7.1 Background

In the original NUREG-1511 (Ref. 2), the NRC staff noted that, as a result of information received in licensee responses to GL 92–01, Revision 1 (Ref. 1), several significant plant-specific issues were identified. These included the PTS evaluations of Palisades and Beaver Valley Unit 1, which showed that these facilities were projected to exceed the PTS screening criteria (refer to 10 CFR 50.61, Ref. 5) before the end of their current operating licenses. Additional information was also cited for Calvert Cliffs Units 1 and 2, Fort Calhoun, Indian Point Unit 3, and Zion Units 1 and 2. In NUREG–1511, Supplement 1 (Ref. 20), the NRC staff provided updated information on Palisades and Calvert Cliffs Units 1 and 2 PTS assessments as well as information on the PTS assessment for Ginna.

This chapter updates the information regarding the PTS assessments for the Palisades and Beaver Valley Unit 1 nuclear plants, and presents new information regarding the staff's structural integrity assessments of plants that have begun the license renewal process. Also included in this section is a brief discussion of an issue that developed with regard to the Fort Calhoun PTS assessment after the staff had completed its review of GL 92–01.

7.2 Palisades

At the time NUREG-1511, Supplement 1, was published, the Consumers Power Company (CPCo, the licensee for the Palisades plant, currently named Consumers Energy) had submitted a revised evaluation of the projected end-of-license (EOL) neutron fluence for the Palisades RPV (Ref. 55). The licensee's analysis was performed by Westinghouse and based upon a combination of physical/geometrical refinements to the neutron transport model, the use of plant-specific dosimetry data, and a specific spectral adjustment routine used in the Westinghouse computer code. The results of the analysis showed a decrease of approximately 25 percent in the EOL fluence for the limiting axial RPV weld. The licensee contended that, based on this reanalysis, the limiting axial welds in the Palisades RPV would remain below the PTS screening criteria through the end of its current operating license.

The NRC issued an interim safety evaluation on this submittal on December 20, 1996 (Ref. 56). With assistance from Brookhaven National Laboratory, the NRC staff concluded that the changes to the projected fluence due the physical/geometrical refinements to the neutron transport code model were acceptable. These changes led to a reduction in the fluence at the RPV of approximately 8 percent. The NRC staff also concluded that there was not sufficient justification to grant the remaining 17 percent reduction requested by the licensee on the basis of dosimetry data and spectral adjustments. The approved 8 percent reduction was sufficient to extend the date at which the limiting axial weld was projected to exceed the PTS screening criteria from 1999 to the end of 2003.

Since December 1996, CPCo and the NRC staff have held additional meetings to discuss whether there is a sufficient basis for granting the 17 percent reduction in the EOL neutron fluence for the Palisades plant. To date, the NRC staff continues to conclude that it is not possible to grant any additional reduction in the projected EOL fluences given the bases which have been presented by the licensee. In early 1999, CPCo did, however, present the NRC with a plan to take a different approach at demonstrating other ways in which a reduction in the projected fluence could be achieved. An initial submittal on this approach was received by the NRC in a letter from CPCo dated March 25, 1999 (Ref 60). After discussing the content of the submittal with the NRC staff, CPCo concluded that the adjustments to the fluence methodology described in the submittal were not likely to be approved without additional justification. CPCo therefore withdrew the submittal of March 25, 1999 until further additional information could be generated. CPCo submitted their most recent fluence analysis for the Palisades RPV on February 21, 2000 (Ref. 61), with the objective of addressing the staff's concerns with the previous fluence methodology. The NRC staff's review of the new fluence methodology is in progress.

7.3 Beaver Valley Unit 1

The Duquesne Light Company (the licensee for the Beaver Valley Nuclear Station) submitted a revised PTS evaluation of the RPV for Beaver Valley Unit 1 to the NRC on August 2, 1996 (Ref. 62). The licensee contended that, as a result of their flux reduction effort and reanalysis of their existing Charpy data (using a hyperbolic tangent curve fitting program), the limiting plate material in the Beaver Valley Unit 1 vessel was now below the PTS screening criteria through the end of the facility's current operating license. Although NRC staff disagreed with certain details of the licensee's evaluation, as cited in NRC's safety evaluation dated October 7, 1997 (Ref. 63), the staff did concur with the licensee's conclusion that the limiting plate in the Beaver Valley Unit 1 RPV was no longer projected to exceed the PTS screening criteria before EOL.

7.4 Fort Calhoun

A neutron fluence evaluation submitted by the Omaha Public Power District, the licensee for Fort Calhoun Station Unit 1, has indicated that the licensee will need to take further action to address PTS. The details of the Fort Calhoun issue are in the letter from L. Raynard Wharton to S. K. Gambhir dated November 30, 1999 (Ref. 64).

7.5 License Renewal Plants: Calvert Cliffs Units 1 and 2, Oconee Units 1, 2, and 3

License renewal applications have been submitted by the licensees for the Calvert Cliffs and Oconee plants (Refs. 65 and 66). The NRC staff has reviewed these applications, in part, with respect to compliance with 10 CFR 50.61 requirements for meeting the pressurized thermal shock screening criteria and the Charpy USE analysis for compliance with Appendix G to 10 CFR Part 50. The staff has issued safety evaluation reports on each application that indicate that the applicable regulations will be satisfied through the license renewal period for each unit (Refs. 67 and 68).

7.6 Reactor Pressure Vessels Fabricated by Multiple Vendors

In the course of recent reviews for several plants, the NRC staff has identified several plants with reactor pressure vessels fabricated by multiple vendors. A description of the circumstances in each case is provided below:

- Browns Ferry Unit 2: the axial welds within each shell course were fabricated by Babcock & Wilcox (B&W), and the circumferential welds were fabricated by Ishikasajima-Hirama Heavy Industries (IHI) of Japan.

- Browns Ferry Unit 3: the axial welds within each shell course were fabricated by B&W, and the circumferential welds were fabricated by IHI.

- Peach Bottom Unit 2: the axial welds within each shell course were fabricated by B&W, and the circumferential welds were fabricated by Chicago Bridge & Iron (CB&I).

- Peach Bottom Unit 3: the axial welds within each shell course were fabricated by B&W, and the circumferential welds were fabricated by CB&I.

- Point Beach Unit 2: one of the beltline circumferential welds (nozzle belt to intermediate shell weld) was fabricated by B&W, and the other circumferential weld (intermediate to lower shell weld) was fabricated by Combustion Engineering (CE).

- Quad Cities Unit 2: the axial welds within each shell course were fabricated by B&W, and the circumferential welds were fabricated by CB&I.

- Surry Unit 1: the axial welds within each shell course and one of the beltline circumferential welds (SA-1650) was fabricated by B&W, and the other beltline circumferential weld (J726) was fabricated by Rotterdam Dockyards.

- Surry Unit 2: the axial welds within each shell course were fabricated by B&W, and the beltline circumferential welds were fabricated by Rotterdam Dockyards.

7.7 Reactor Pressure Vessels Fabricated from Forging Materials

Approximately 20 RPVs for US nuclear power plants (all PWRs) were fabricated from forgings. Although most of these forged RPVs were fabricated from SA-508 Class 2 steel, the RPVs for Braidwood Units 1 and 2 and Prairie Island Units 1 and 2 were fabricated from SA-508 Class 3 steel. A listing of the plants with forged RPVs and the forging fabricator is provided below:

- Braidwood 1 - Japan Steel Works
- Braidwood 2 - Japan Steel Works
- Byron 1- Ladish
- Byron 2 - Japan Steel Works
- Catawba 1 - Rotterdam Dockyard
- Davis-Besse - Ladish
- Ginna - Bethlehem Steel
- Kewaunee - Bethlehem Steel
- McGuire 2 - Rotterdam Dockyard
- North Anna 1 - Rotterdam Dockyard
- North Anna 2 - Rotterdam Dockyard
- Oconee 2 - Ladish
- Oconee 3 - Ladish
- Point Beach 2 - Bethlehem Steel
- Prairie Island 1 - Creusot-Loire
- Prairie Island 2 - Creusot-Loire
- Sequoyah 1- Rotterdam Dockyard
- Sequoyah 2- Rotterdam Dockyard
- Turkey Point 3 - Bethlehem Steel
- Turkey Point 4 - Bethlehem Steel
- Watts Bar - Rotterdam Dockyard

8 SUMMARY AND CONCLUSIONS

Since the issuance of GL 92–01, Revision 1 (Ref. 1), in March 1992, and NUREG-1511 (Ref. 2), in December 1994, the staff has directed its efforts toward determining the generic implications of the larger-than-expected variability observed in the chemical composition of RPV welds. To address this concern, the staff issued Supplement 1 to GL 92–01, Revision 1 (Ref. 3), on May 18, 1995, and requested that the licensees to which the GL was addressed identify any new data which could be pertinent to and impact the structural integrity analyses of their RPVs relative to the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR Part 50, Appendices G and H (Refs. 4-7).

The majority of the licensees responding to GL 92–01, Revision 1, Supplement 1, indicated that they were participating in the activities of their respective industry owners group (e.g., either the BWRVIP, CEOG, or B&WOG) to collect and analyze available RPV chemistry (specifically, copper and nickel content data) and surveillance data on behalf of its member utilities. The owners groups have now completed these initiatives. This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light-water nuclear power plants. These efforts should minimize surprises regarding the alloying chemistries for domestic RPV beltline materials. However, the staff expects additional surveillance data will become available after fracture toughness testing is performed on surveillance capsules that are presently being irradiated in domestic RPVs. The staff will incorporate the additional surveillance data into the existing database after it becomes available for review.

The review of the industry's vessel integrity initiatives led the staff to conclude that the new data compiled by the owners groups could have an impact on PTS and P-T limit assessments for some facilities. Therefore, from summer to fall of 1998, the staff issued RAIs to the majority of licensees that responded to GL 92–01, Revision 1, Supplement 1, and requested that the recipients assess the impact of the newly-compiled chemistry and surveillance data, if any, on the PTS and P–T limit evaluations for their facilities. The staff reviewed the licensee responses to the RAIs and all licensee responses to GL 92–01 Revision 1, and GL 92–01, Revision 1, Supplement 1, along with the applicable reports from the owners groups, to determine the appropriate chemical composition and neutron embrittlement trends for each RPV. In some cases, these reviews indicated the need for additional discussions with the licensee to clarify or justify assumptions and conclusions made in the submittals. The results of these reviews have been used to prepare revisions to the data contained in the RVID for each plant.

Several developments have also occurred in the areas of Codes and Standards activities and research developments that affect RPV integrity assessments. In particular, the ASME Code Committees have passed Code Cases N–640 and N–588 (Refs. 9 and 10), which have direct consequences on P–T limit assessments, and Code Case N–629 (Ref. 11), which proposes a new method (the "Master Curve") for indexing material reference temperatures. Research developments have included work on new RPV embrittlement correlations, RPV flaw distribution studies, ongoing work to support the Master Curve methodology, and an effort by the NRC and industry to revise the PTS screening criteria.

The staff has also considered proposals to reduce the scope of augmented inspections performed on the shell welds of BWR RPVs. The basis for this proposed reduction in scope for the augmented inspections, which are required pursuant to Paragraph (g)(6)(ii)(A) of 10 CFR 50.55a, was proposed by the BWRVIP in Topical Report BWRVIP–05. The staff approved this topical report in its SER dated July 28, 1998 (Ref. 14) , and informed the industry of its position to allow reduced inspection scopes for inspections of circumferential shell welds in BWR RPVs in GL 98-05, which was issued on November 11, 1998 (Ref. 47). The staff has currently approved a number of proposals for reduced inspections of the circumferential RPV shell welds in BWR-designed facilities.

With the adoption of 10 CFR 50.66 (Ref. 16), the staff has also established a regulatory framework for thermal annealing of RPVs. The staff's regulatory position on thermal annealing is documented in RG 1.162 (Ref. 17).

The feasibility of thermal annealing was demonstrated by a DOE/Industry–sponsored annealing demonstration project (ADP); this ADP was performed at a decommissioned Marble Hill reactor unit, and applied indirect heating as the method for annealing the unit's RPV. A second ADP using an electrical resistance heating approach was canceled. Currently, no licensee is planning to anneal the RPV of any U.S. facility.

The staff has also updated the RVID. RVID 2 was issued on the World Wide Web in June 1999. RVID 2 is a Windows 3.1 native application based on Microsoft Access 2.0™. The four RVID 2 diskettes can be downloaded from the homepage at (http://www.nrc.gov/NRR/RVID/index.html) which is linked to the NRC homepage. The RVID was developed following the staff's review of licensee responses to GL 92–01, Revision 1. The database summarizes the properties of the reactor vessel beltline materials for each operating commercial nuclear power plant. RVID 2 reflects license responses to the RAIs on GL 92–01, Revision 1, Supplement 1. The database will be updated when sufficient amounts of new surveillance data, chemistry data, or fluence evaluations warrant a new revision.

9 REFERENCES

1. GL 92–01, Revision 1, "Reactor Vessel Structural Integrity," March 6, 1992.

2. NUREG-1511, "Reactor Pressure Vessel Status Report," December 1994.

3. Generic Letter 92–01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 18, 1995.

4. Section 50.60 to Part 50 of Title 10 to the *Code of Federal Regulations*, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation."

5. Section 50.61 to Part 50 of Title 10 to the *Code of Federal Regulations*, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

6. Appendix G to Part 50 of Title 10 to the *Code of Federal Regulations*, "Fracture Toughness Requirements."

7. Appendix H to Part 50 of Title 10 to the *Code of Federal Regulations*, "Reactor Vessel Material Surveillance Program Requirements."

8. SECY-95-119, "Status of Reactor Pressure Vessel Issues," May 8, 1995.

9. American Society for Mechanical Engineers Code Case N-640, "Alternative to Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I," September 18, 1998.

10. American Society for Mechanical Engineers Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," December 12, 1997.

11. American Society for Mechanical Engineers Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials for Vessels of Section III, Division 1, Class 1," April 1998.

12. Section 50.55a to Part 50 of Title 10 to the *Code of Federal Regulations*, "Codes and Standards."

13. BWR Vessel and Internals Project Topical Report No. EPRI TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," September 1995.

14. Letter from G. C. Lainas, Acting Director - Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to C. T. Terry, Chairman - Boiling Water Reactor Vessel and Internals Project, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," July 28, 1998.

15. Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998.

16. Section 50.66 to Part 50 of Title 10 to the *Code of Federal Regulations*, "Requirements for Thermal Annealing of the Reactor Pressure Vessel."

17. Regulatory Guide 1.162, "Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels," February 1996.

18. Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, 1995 Edition through 1996 Addenda.

19. NUREG/CR-6023, "Generic Analyses for Evaluation of Low Charpy USE Effects on Safety Margins Against Fracture of RPV Materials," Oakridge National Laboratory, July 1993.

20. NUREG-1511, Supplement 1, "Reactor Pressure Vessel Status Report," October 1996.

21. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

22. CE NPSD-1039, Revision 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," ABB Combustion Engineering Nuclear Operations, Windsor, CT, June 1997.

23. CE NPSD-1119, Revision 1, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content," ABB Combustion Engineering Nuclear Operations, Windsor, CT, July 1998.

24. BAW-2325, Supplement 1, "Supplement to the Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," Framatome Technologies, Lynchburg, VA, December 1998.

25. BAW-2325, Revision 1, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," Framatome Technologies, Lynchburg, VA, January 1999.

26. EPRI TR-109727, "BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," Electric Power Research Institute, Palo, Alto, CA, December 1997.

27. Memorandum from K. R. Wichman to E. J. Sullivan, "Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92–01, Revision 1, Supplement 1 Responses," November 19, 1997.

28. "NRC/Industry Workshop on Reactor Pressure Vessel Integrity, February 12 and 13, 1998," Nuclear Energy Institute, Washington, D. C.

29. Letter from G. C. Lainas, Acting Director - Division of Engineering, U.S. Nuclear Regulatory Commission, to R. O. Hardies, Baltimore Gas and Electric Company, "Combustion Engineering Owners Group Report CE NPSD-1039 Revision 2, 'Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds' and CE NPSD-1039 Appendix A, Revision 2, 'CE Reactor Vessel Weld Properties Database Volumes 1 and 2'," March 27, 1998.

30. Letter from S. C. Black, Deputy Director - Division of Licensing Program Management, U.S. Nuclear Regulatory Commission, to E. R. Kane, Framatome Services, "NRC Inspection of Framatome Technologies, Inc. (Inspection Report No.: 99901300/97-01)," January 28, 1998.

31. Letter from G. C. Lainas, Acting Director - Division of Engineering, U.S. Nuclear Regulatory Commission, to C. D. Terry, Vice President - Nuclear Safety Assessment and Support, Niagra Mohawk Power Company, "Closeout for BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," March 27, 1998.

32. American Society for Testing and Materials E 900, "Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)."

33. NUREG/CR-6551, "Improved Embrittlement Correlations for Reactor Pressure Vessel Steels," November 1998

34. Marshall Committee, "An Assessment of the Integrity of PWR Pressure Vessels," United Kingdom Atomic Energy Authority, October 1, 1976.

35. NUREG/CR-6471, Vol. 1, PNNL-11143, "Characterization of Flaws in U.S. Reactor Pressure Vessels, Density and Distribution of Flaw Indications in PVRUF," November 1998.

36. NUREG/CR-6471, Vol. 3, PNNL-11143, "Characterization of Flaws in U.S. Reactor Pressure Vessels, Density and Distribution of Flaw Indications in the Shoreham Vessel," November 1999.

37. Wallin, K., "Irradiation Damage Effects on the Fracture Toughness Transition Curve Shape for Reactor Vessel Steels," *International Journal of Pressure Vessels & Piping,* **55**, pp. 61-79, 1993.

38. American Society for Testing and Materials Standard Procedure E 1921-97, "Standard Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range," February 1998.

39. Letter NRC-99-039 from M. L. Marchi, Vice President - Nuclear, Wisconsin Public Service Company, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Proposed Technical Specification Change Regarding Heatup and Cooldown Limit Curves and Request for Exemption to 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to Part 50," June 7, 1999.

40. Transcript of the Commission Meeting dated May 12, 1999.

41. Commission Staff Requirements Memorandum No. SRM M970512B, May 30, 1997.

42. Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05," August 7, 1997.

43. Letter from the U.S. Nuclear Regulatory Commission to the Boiling Water Reactor Vessel and Internals Project, "Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information," August 14, 1997.

44. Information Notice 97-63, Supplement 1, "Status of NRC Staff's Review of BWRVIP-05," May 7, 1998.

45. NUREG-1560, Volume 1, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997.

46. Regulatory Guide 1.154, "Format and Content of Plant-specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.

47. Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Inspection Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 11, 1998.

48. Letter from Mr. V. Wagoner, Technical Chairman, BWRVIP Integration Committee to the U.S. Nuclear Regulatory Commission Document Control Desk, "Project No. 704 Response to NRC Comments on BWRVIP Analyses of BWR Reactor Pressure Vessel Axial Welds," November 12, 1999.

49. Letter from J. R. Strosnider, Director of Engineering, U.S. Nuclear Regulatory Commission, to C. Terry, BWRVIP Chairman, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," March 7, 2000.

50. American Society for Testing and Materials, "Recommended Guide for In-Service Annealing of Water-Cooled Nuclear Reactor Vessels," ASTM E 509-86, Philadelphia, 1986.

51. ASME Code Case N-557, "In Place Dry Annealing of a PWR Nuclear Reactor Vessel," March 1996.

52. Letter from R. W. Smedley, Manager-Licensing, Consumers Power Company, to the U.S. Nuclear Regulatory Commission, "Docket 50-255-License DPR-20-Palisades Plant 10CFR50.61-Preliminary Thermal Annealing Report Section 3," October 12, 1995.

53. - Letter from T. C. Bordine, Manager - Licensing, Consumers Power Company, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Response to Request for Additional Information Regarding the Preliminary Thermal Annealing Report.," August 26, 1996.

54. Letter from T. C. Bordine, Manager - Licensing, Consumers Power Company, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Preliminary Thermal Annealing Report - Withdrawal of Request to Review," April 24, 1997.

55. Letter from R. W. Smedley, Manager-Licensing, Palisades Plant, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Docket 50-255, License DPR-20, Palisades Plant Updated Reactor Vessel Fluence Values, April 4, 1996.

56. Letter from J. N. Hannon, Director - Project Directorate III-1, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to T. C. Bordine, Manager - Licensing, Consumers Power Company, "Palisades: Evaluation of Updated Reactor Pressure Vessel Fluence Values," December 20, 1996.

57. Letter from J. M. Taylor, Executive Director for Operations, U.S. Nuclear Regulatory Commission, to Dr. T. R. Lash, Director - Office of Nuclear Energy, Science and Technology, U.S. Department of Energy, enclosing Two Copies of the Memorandum of Understanding for the DOE Annealing Demonstration Project, August 4, 1996.

58. Electric Power Research Institute Report TR-108316, "Marble Hill Thermal Annealing Demonstration Project," March 1998.

59. NUREG/CR-6552, "Marble Hill Annealing Demonstration Evaluation," February 1998.

60. Letter from N. L. Haskell, Director of Licensing, Consumers Energy Company) to the U.S. Nuclear Regulatory Commission Document Control Desk, "Palisades Plant, Revised Best Estimate Fluence Evaluation Using Industry Data," March 25, 1999.

61. Letter from D. G. Malone, Acting Director of Licensing, Consumers Energy, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Palisades Reactor Vessel Neutron Fluence Reevaluation," February 21, 2000.

62. Letter from S. C. Jain, Vice President - Nuclear Services, Duquesne Light Company, to the US Nuclear Regulatory Commission Document Control Desk, "Beaver Valley Power Station Unit No.1, 10 CFR 50.61 RT-PTS Submittal," August 2, 1996.

63. Letter from D. S. Brinkman, Project Manager, U.S. Nuclear Regulatory Commission, to J. E. Cross, Chief Nuclear Officer, Duquesne Light Company, "Safety Evaluation Regarding Pressurized Thermal Shock (PTS) Assessment for Beaver Valley Power Station Unit No. 1 . . . ," October 7, 1997.

64. Letter from L. R. Wharton, Project Manager, U.S. Nuclear Regulatory Commission, to S. K. Gambir, Division Manager of Nuclear Operations, Omaha Public Power District, "Concerns with Fort Calhoun Station Basis for Prediction of RT_{PTS} at Expiration of License . . . ," November 30, 1999.

65. Letter from C. H. Cruse, Vice President - Nuclear Energy, Baltimore Gas and Electric Company, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Application for License Renewal," April 8, 1998.

66. Letter from M. S. Tuckman, Executive Vice President - Nuclear Generation, Duke Power Company, to the U.S. Nuclear Regulatory Commission Document Control Desk, "Oconee Nuclear Station, Unit 1, 2, and 3, ... Application for Renewed Operating Licenses," July 6, 1999.

67. NUREG-1705, "Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2," December 1999.

68. NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," March 2000.

APPENDIX

PTS TABLES FOR PRESSURIZED WATER REACTORS AND LIMITING MATERIAL TABLES FOR BOILING WATER REACTORS

APPENDIX - PWR RTpts VALUES FOR LIMITING MATERIALS (RTpts ONLY APPLICABLE TO PWRs)

PLANT NAME	LIMITING MATERIAL BELTLINE ID	HEAT ID	RTpts @ EOL	SCREENING CRITERIA
ARKANSAS NUCLEAR 1	UPPER/LOWER SHELL CIRC. WELD WF-112	406L44	237	300
ARKANSAS NUCLEAR 2	LOWER SHELL C-8010-1	C-8161-2	123	270
BEAVER VALLEY 1	LOWER SHELL B6903-1	C6317-1	268	270
BEAVER VALLEY 2	INTERMEDIATE SHELL B9004-1	C0544-1	153	270
BRAIDWOOD 1	LOWER NOZZLE BELT FORGING	5P-7016	52	270
BRAIDWOOD 2	LOWER NOZZLE BELT FORGING	5P-7056	70	270
BYRON 1	INT. SHELL FORGING	5P-5933	111	270
BYRON 2	MIDDLE CIRC. WELD WF-447	442002	101	300
CALLAWAY	LOWER SHELL R2708-3	C4499-1	115	270
CALVERT CLIFFS 1	LOWER SHELL AXIAL WELD 3-203A/C	21935	240	270
CALVERT CLIFFS 2	INTERMEDIATE SHELL D-8906-1	A-4463-1	189	270
CATAWBA 1	LOWER SHELL 04 FORGING	527708	60	270
CATAWBA 2	INTERMEDIATE SHELL B8605-2	C0543-2	130	270
COMANCHE PEAK 1	LOWER SHELL R1108-1	C4464-1	100	270
COMANCHE PEAK 2	INT SHELL A3807-2	C5522-2	92	270
COOK 1	INT. SHELL AXIAL WELDS 2-442 A,B,&C	13253/12008(T)	215	270
COOK 2	INTERMEDIATE SHELL PLATE 10-1	C5556-2	216	270
CRYSTAL RIVER 3	LOWER SHELL AXIAL WELDS SA-1580	8T1762	211	270
DAVIS-BESSE	UPPER/LOWER SHELL CIRC WELD WF-182-1	821T44	191	300
DIABLO CANYON 1	LOWER SHELL AXIAL WELD 3-442C	27204	258	270
DIABLO CANYON 2	INTERMEDIATE SHELL B5454-2	C5168-2	211	270
FARLEY 1	LOWER SHELL B6919-1	C6940-1	183	270
FARLEY 2	INTERMEDIATE SHELL B7212-1	C7466-1	205	270
FORT CALHOUN	LOWER SHELL AXIAL WELDS 3-410A/C	27204/12008(T)	268	270
GINNA	INT./LOWER SHELL CIRC. WELD SA-847	61782	254	300
INDIAN POINT 2	INTERMEDIATE SHELL B2002-3	B-4922-1	230	270
INDIAN POINT 3	LOWER SHELL B2803-3	A-0512-2	265	270
KEWAUNEE	INT./LOWER CIRC. WELD	1P3571	277	300
MCGUIRE 1	LOWER SHELL AXIAL WELD M1.32	21935/12008	231	270
MCGUIRE 2	LOWER SHELL 04	411337-11	141	270
MILLSTONE 2	LOWER SHELL C-506-1	C-5667-1	177	270
MILLSTONE 3	INTERMEDIATE SHELL B9805-1	C-4039-2	134	270
NORTH ANNA 1	LOWER SHELL FORGING 03	990400/292332	184	270
NORTH ANNA 2	LOWER SHELL 03	990533/297355	220	270
OCONEE 1	INTERMEDIATE SHELL AXIAL WELDS SA-1073	1P0962	214	270
OCONEE 2	MIDDLE CIRC WELD WF-25	299L44	273	300
OCONEE 3	UPPER/LOWER SHL CIRC WELD (INSIDE 75%) WF-67	72442	236	300
PALISADES	LOWER SHELL AXIAL WELDS 3-112A/C	W5214	269	270
PALO VERDE 1	INTERMEDIATE SHELL M-6701-3	C4188-1	123	270

APPENDIX - PWR RTpts VALUES FOR LIMITING MATERIALS (RTpts ONLY APPLICABLE TO PWRs)

PLANT NAME	LIMITING MATERIAL BELTLINE ID	HEAT ID	RTpts @ EOL	SCREENING CRITERIA
PALO VERDE 2	INTERMEDIATE SHELL F-765-6	63716-1	78	270
PALO VERDE 3	LOWER SHELL F-6411-2	79745-1	68	270
POINT BEACH 1	CIRCUMFERENTIAL WELD SA-1101	71249	274	300
POINT BEACH 2	INTERMEDIATE TO LOWER SHELL CIRC. WELD SA1484	72442	288	300
PRAIRIE ISLAND 1	NOZZLE TO INT. SHELL CIRC. WELD	2269	163	300
PRAIRIE ISLAND 2	NOZZLE SHELL TO INTERMEDIATE SHELL CIRC. WELD	1752	150	270
ROBINSON 2	UPPER CIRC WELD 10-273	W5214	255	300
SALEM 1	LOWER SHELL AXIAL WELDS 3-042 C	34B009	253	270
SALEM 2	LOWER SHELL AXIAL WELDS 3-442 A&C	21935/12008(T)	227	270
SEABROOK	LOWER SHELL R1808-1	D1081-3	120	270
SEQUOYAH 1	CIRC WELD	25295	235	300
SEQUOYAH 2	INTERMEDIATE SHELL FORGING 05	288757/981057	152	270
SHEARON HARRIS	INTERMEDIATE SHELL	B4197-2	196	270
SONGS-2	LOWER SHELL C-6404-5	C-7585-1	146	270
SONGS-3	INTERMEDIATE SHELL C-6802-1	C-9195-2	125	270
SOUTH TEXAS 1	INTERMEDIATE SHELL R1606-3	C-4326-2	84	270
SOUTH TEXAS 2	INTERMEDIATE SHELL R2507-1	NR 62 067-1	67	270
ST. LUCIE 1	LOWER SHELL AXIAL WELDS 3-203	305424	206	270
ST. LUCIE 2	INTERMEDIATE SHELL M-605-2	B-3416-2	163	270
SUMMER	LOWER SHELL	C9923-1	113	270
SURRY 1	LOWER SHELL AXIAL WELDS L2 SA-1526	299L44	245	270
SURRY 2	INT. TO LOWER SHELL CIRC. WELD R3008	0227 RDAM	215	300
TMI-1	LOWER SHELL AXIAL WELD (100%) (SA-1526)	299L44	262	270
TURKEY POINT 3	INT. TO LOWER SHELL CIRC WELD SA-1101	71249	279	300
TURKEY POINT 4	INT./LOWER SHELL CIRC WELD SA-1101	71249	279	300
VOGTLE 1	INTERMEDIATE SHELL B8805-2	C-0613-2	118	270
VOGTLE 2	LOWER SHELL R8-1	C-4304-1	126	270
WATERFORD 3	LOWER SHELL M-1004-2	57286-1	76	270
WATTS BAR 1	INTERMEDIATE SHELL 05	527536	253	270
WOLF CREEK	LOWER SHELL R2508-1	B8759-2	104	270
ZION 1	MIDDLE CIRC WELD WF-70	72105	258	300
ZION 2	INT./LOWER SHELL CIRC WELD SA-1769	71249	272	300

NOTES:

1.) VALUES APPLY TO EXISTING LICENSE, AND WILL NEED TO BE RE-ASSESSED FOR LICENSE RENEWAL

2.) FOR UPPER SHELF ENERGY INFORMATION, REFER TO NUREG 1511 (DECEMBER 1994)

APPENDIX - BWR LIMITING MATERIALS

PLANT NAME	LIMITING MATERIAL BELTLINE ID	HEAT ID	ART @ EOL
BROWNS FERRY 1	CIRC WELD WF-154	406L44	147
BROWNS FERRY 2	AXIAL WELDS	NA/W-A	144
BROWNS FERRY 3	AXIAL WELDS ES	NA/W-A	144
BRUNSWICK 1	NOZZLE FORGING N16A	Q2Q1VW	119
BRUNSWICK 2	NOZZLE FORGING N16B	Q2Q1VW	110
CLINTON	WELDS	76492	147
COOPER	LOWER INTERMEDIATE SHELL G-2802-2	C2307-2	160
DRESDEN 2	LOWER SHELL AXIAL WELD	PQ1092C-2	100
DRESDEN 3	LOWER INT. TO LOWER SHELL CIRC WELD	299L44/8650	129
DUANE ARNOLD	LOWER INTERMEDIATE SHELL 1-20	B0436-2	145
FERMI 2	LOWER SHELL AXIAL WELDS 2-307A,B,C	13253/12008	87
FITZPATRICK	LOWER SHELL AXIAL WELDS 2-233A/C	27204/12008	128
GRAND GULF 1	#2 SHELL AXIAL WELDS	627260	72
HATCH 1	LOWER INT. SHELL G-4804-2	C4114-2	152
HATCH 2	LOWER SHELL G6603-2	C8553-1	74
HOPE CREEK	INT-LOWER TO INT. SHELL CIRCUMFERENTIAL WELD	D53040	14
LASALLE 1	MIDDLE SHELL AXIAL WELDS 3-308 A/C	1P3571	93
LASALLE 2	LOWER SHELL 21-2	C9425-1	86
LIMERICK 1	SHELL COURSE # 2 17-2	C7677-1	95
LIMERICK 2	SHELL COURSE # 1 14-2	B3416-1	130
MONTICELLO	LOWER/INT. SHELL I-14	C2220-1	165
NINE MILE POINT 1	UPPER SHELL G-307-4	P2076	173
NINE MILE POINT 2	NUMBER 2 SHELL	C3147-1	59
OYSTER CREEK	LOWER-INT. SHELL G-8-6	P2150-1	174
PEACH BOTTOM 2	LOWER INTERMEDIATE SHELL	C2873-1	59
PEACH BOTTOM 3	LOWER INTERMEDIATE SHELL 6-139-10	C2773-2	83
PERRY	AXIAL WELD	627260/B322A27AE	92
PILGRIM	LOWER INT. SHELL AXIAL WELDS 1-338A,B,C	27204/12008	120
QUAD CITIES 1	LOWER AND LOWER INTERMEDIATE AXIAL WELDS	PQ1300	99
QUAD CITIES 2	LOWER INTERMEDIATE & LOWER SHELL AXIAL WELDS	PQ1300	112
RIVER BEND	AXIAL WELDS	5P6756/0342(T)	114
SUSQUEHANNA 1	LOWER INTERMEDIATE SHELL 22-3	C2433-1	66
SUSQUEHANNA 2	LOWER INTERMEDIATE SHELL 22-1	C2421-3	58
VERMONT YANKEE	LOCATION UNKNOWN 1-15	C3116-2	68
WNP-2	#2 RING	B5301-1	49

NOTES:

1.) VALUES APPLY TO EXISTING LICENSE, AND WILL NEED TO BE RE-ASSESSED FOR LICENSE RENEWAL

2.) FOR UPPER SHELF ENERGY INFORMATION, REFER TO NUREG 1511 (DECEMBER 1994)

3.) HEAT ID's BEGINNING WITH "NA" ARE UNKNOWN (ELECTROSLAG WELDS)

4.) ART @ EOL IS THE VALUE MEASURED AT PEAK FLUENCE AT THE INSIDE SURFACE

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

(Condensed Version of the Abstract from the Report)

On May 18, 1995, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," requesting that addressees: (1) identify, collect, and report any new data pertinent to the analysis of structural integrity for the reactor pressure vessels (RPVs) at their nuclear power plants, and (2) assess the impact of those data on their RPV integrity analyses relative to the requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and to the requirements of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements". Since the issuance of GL 92-01, Revision 1, Supplement 1, the industry completed a major initiative to collect all available alloying chemistry and materials property data for the various forging, plate and weld materials used to fabricate the RPVs. This represents the first time that a comprehensive, integrated assessment of all alloying chemistry and surveillance data has been completed for the materials that have been used to fabricate the RPVs in U.S. light- water nuclear power plants. In the summer and fall of 1998, the staff issued a series of requests for additional information (RAIs) regarding the industry's responses to GL 92-01, Revision 1, Supplement 1. In the RAIs, the staff requested that the addressees assess how the updated alloying chemistry and materials property data would affect the results of the RPV integrity analyses for their plants. This report, in part, summarizes both the industry's and the NRC's efforts to address how all of the new chemistry and surveillance data, when integrated, could affect the plant-specific RPV integrity analyses for the RPVs of U.S. light-water nuclear power plants.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

10 CFR 50.61; 10 CFR Part 50, Appendix G; 10 CFR Part 50, Appendix H; pressurized thermal shock (PTS); upper shelf energy (USE); reactor pressure vessel (RPV); material property surveillance data; Reactor Vessel Integrity Database (RVID); best estimate chemistry; adjusted reference temperature (ART or RTndt); adjusted reference temperature for pressurized thermal shock (RTpts); initial RTndt (RTndt(u)); base metal materials (plates or forging materials); longitudinal (axial) weld materials; circumferential (girth) weld materials; screening criteria for pressurized thermal shock; reactor vessel annealing

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